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APPENDIX A

Code Summary

CASMO-2

CASMO-2 is a multi-group two-dimensional transport theory code for burnup calculations on BWR and PWR assemblies. This code has been developed by Studsvik Energiteknik AB and supported by EPRI.

CHART

CHART prepares cross section tables in HARMONY format from cross section data produced by CASMO-2. CHART reduces significantly the tedious task of hand transferring values from CASMO-2 printout to macroscopic and microscopic tables in card image HARMONY format. Two, three, and four group cross section data may be obtained with one-dimensional HARMONY interpolating tables.

CORE

CORE (Codes for Operating Reactor Evaluation), is a package of computer routines for the off-line evaluation of reactor performance. CORE uses as input: detailed reactor physics data, isotopics, and thermal-hydraulics data. Calculated values are: F_Q, $F^N_{\Delta H}$, assembly burnups, isotopics, reactivity, and core thermal-hydraulics information.

DELAY

DELAY calculates core averaged delayed neutron fractions for six energy groups, core averaged decay constants for six energy groups, core averaged delayed neutron fraction with and without importance factor, estimated prompt neutron lifetime, and reactivity versus period. Input consists primarily of isotopic fission fractions versus burnup and enrichment from PDQ07 calculations.

EPRI-CELL

EPRI-CELL computes the space, energy and burnup dependence of the neutron spectrum within cylindrical cells of Light Water Reactor fuel rods. Its primary output consists of broad group, microscopic, exposure dependent cross sections for subsequent use in multidimensional diffusion theory depletion analysis. EPRI-CELL utilizes three industry accepted subcodes; GAM-1, THERMOS, and CINDER.

EPRI-CPM

EPRI-CPM is a multi-group two-dimensional collision probability code for burnup calculations on BWR and PWR assemblies. The code handles a geometry consisting of cylindrical fuel rods of varying composition in a square pitch array with allowance for fuel rods loaded with gadolinium, burnable absorber rods, cluster control rods, in-core instrument channels, water gaps, boron steel curtains and cruciform control rods in the regions separating fuel assemblies.

EPRI-FIT

EPRI-FIT is a program which processes the PDQ07 integral file and calculates and edits values needed by the EPRI-NODE code. EPRI-FIT greatly reduces the hand calculation time needed to extract these values from the PDQ07 printout and improves the quality assurance. A data file under the local name of COLOR is written which contains the EPRI-FIT edited data and is used as input to the SUPERLINK program.

EPRI-NODE

EPRI-NODE is a multi-dimensional nodal code derived from FLARE. The EPRI-NODE program computes the core effective multiplication factor, the three-dimensional core power distribution, core coolant flow and temperature distribution, and fuel exposure distribution. The program includes the effects of partially inserted full-length control rods, part-length rods, and up to 13 different fuel assembly types with different enrichments and burnable absorber shim loadings. EPRI-NODE has a capacity to represent the core with 32 axial nodes for each fuel assembly and 30x30 nodes in the XY plane.

The program iterates to account for the interaction between power distribution and core nuclear properties which depend on coolant flow and coolant temperature distributions, fuel temperature distribution and xenon distribution. The program computes the time dependence of xenon following changes in power level and/or changes in power distribution. The program permits fuel shuffling from one location to another and fresh fuel insertion for burnup cycle calculations. Individual steps can by stacked for either xenon transient or fuel cycle burnup calculations. See Reference 5.

EPRI-NUPUNCHER

NUPUNCHER prepares cross section tables in HARMONY format from cross section data produced by EPRI-CELL and placed on the ECDATA file. NUPUNCHER reduces significantly the tedious task of hand transferring values from the EPRI-CELL printout to macroscopic and microscopic tables in card image HARMONY format. Two, three and four group cross section data may be obtained with one-dimensional HARMONY interpolating tables.

EPRI-PDQ07 MODIFICATIONS

PDQ07 is an industry accepted multi-group one, two, or three-dimensional diffusion depletion code. EPRI-ARMP uses PDQ07/Version II with minor modifications to allow options for improved removal treatment, peak power editing, and re-editing.

EPRI -SHUFFLE

The EPRI-SHUFFLE program will read a PDQ07 concentration file, make certain modifications to this file, and write a new updated concentration file. This procedure is accomplished by defining "assembly regions" in the program input. Assembly regions are square arrays of mesh points containing depletable nuclide concentrations and superimposed on the original PDQ07 geometry. These assembly regions are then used to describe the movement of existing nuclide concentrations by translation, reflection and/or rotation. In addition, new fuel concentrations can replace spent fuel concentrations in selected assembly regions described in the program's input.

EPRI - SUPERLINK

SUPERLINK accesses data on the files produced by EPRI-FIT and, together with relevant input information for file management and for data processing control, produces polynomial coefficients for use in EPRI-NODE.

MULTIFIT

MULTIFIT reads EPRI-CELL cross section files and generates HARMONY cross sections and g-factors. Both HARMONY masks and function tables can include the effects of up to three independent variables. MULTIFIT can perform almost all of the functions of EPRI-NUPUNCHER.

PDQ07

See EPRI-PDQ07 Modifications and Reference 4.

CASMO-3

CASMO-3 is a multi-group two-dimensional transport theory code for burnup calculations on BWR and PWR assemblies. This code develops cross-section data for use in SIMULATE-3. A full description of this code is contained in Reference 28.

SIMULATE-3P

SIMULATE-3 is a three-dimensional, two-group diffusion theory reactor simulator used for nuclear design calculations. A full description of this code is contained in Reference 28.

APPENDIX B

NRC/DPC Correspondence Regarding NRC Request for Additional Information

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

November 5, 1984

Docket Nos: 50-369, 50-370
and 50-413, 50-414

Mr. H. B. Tucker, Vice President
Nuclear Production Department
Duke Power Company
422 South Church Street
Charlotte, North Carolina 28242

Dear Mr. Tucker:

Subject: Request for Additional Information Regarding Topical Report on Physics Methodology for Reloads: McGuire and Catawba Nuclear Station

In response to your letter of July 18, 1984, the NRC staff, with the technical assistance of Brookhaven National Laboratory (BNL), is reviewing Duke Power Company topical report DPC-NF-2010 which describes the nuclear physics methodology for reload design at the McGuire and Catawba Nuclear Stations. We find that additional information identified in the enclosure is needed to complete this review.

A reply at your earliest opportunity and no later than November 30, 1984, is needed for the staff to meet your requested review completion date of January 1985. A copy of your reply should also be forwarded directly to BNL at the address below.

Should you have questions or need to meet with the staff regarding the enclosure, contact Darl S. Hood at (301) 492-8408.

Sincerely,

Elinor G. Adensam

Elinor G. Adensam, Chief
Licensing Branch No. 4
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Enclosure:
As stated

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See next page

CATAWBA

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ENCLOSURE

REQUEST FOR ADDITIONAL INFORMATION ON DUKE POWER COMPANY TOPICAL REPORT DPC-NF-2010

1. Please provide additional information regarding the NUC-MARGINS code and its use in the Dropped Rod Analysis. Provide short descriptions of the input, output, calculational models used, benchmark calculations performed and the conservatisms assumed in the analysis.

2. Identify the nominal and various off-nominal cross-section sets that are generated in order to evaluate the different reactivity coefficients and defects.

3. Provide a short description of the PDQEDIT code and describe the verification program that was undertaken to test data generated with PDQEDIT for use in SNA-CORE.

4. Comment on the reasons for the 3.1% non-conservative bias in the calculated peak axial powers (Section 11.5.4). Describe the model refinements, if any, that have been undertaken to reduce this bias.

5. Duke Power Company's contention that no uncertainty in calculated pin powers needs to be accounted for has not been adequately established. One possible way to establish the uncertainty is to perform a standard problem. A standard problem recently developed at Brookhaven National Laboratory for a licensee to assess its ability to calculate typical PWR fuel assemblies, is attached. A solution of this problem or other justification for the assumed uncertainty should be provided.

6. Please provide the updates to DPC-NF-2010, if any, that will make it consistent with the methodologies currently being used by Duke Power.

FUEL ASSEMBLY STANDARD PROBLEM

The standard problem is to be calculated in two dimensions in an iterated-source mode using reflecting boundary conditions in the horizontal plane neglecting axial leakage. The following series of assembly depletion and reactivity defect calculations are to be calculated.

I. DEPLETION CALCULATIONS

Provide the following edited quantities for an assembly with and without burnable poison rods at BOL, 500, 5000, 10000, 20000, 30000 and 40000 Mwd/MT*:

1. Relative pin powers

2. Assembly volume averaged fuel pellet isotopics; U^{235}, U^{238}, Pu^{239}, Pu^{240}, Pu^{241}, Pu^{242} and calculated fission product densities [atom/barn-cm]

3. Assembly total reaction rates (A-absorption, F-fission)

   a. Fuel

U^{235} (A)	Pu^{240} (A)
U^{235} (F)	Pu^{240} (F)
U^{238} (A)	Pu^{241} (A)
U^{238} (F)	Pu^{241} (F)
Pu^{239} (A)	Pu^{242} (A)
Pu^{239} (F)	Pu^{242} (F)

   b. Clad (A)
   c. Burnable Poison (A)
   d. Water (A)
   e. Control Rod (A)

4. Assembly Characteristics

   a. k_∞ - Infinite Multiplication Factor
   b. M^2 - Migration Area [cm^2]
   c. B_M^2 - Material Buckling [cm^{-2}]
   d. β - Delayed Neutron Fraction
   e. Two-Group Inverse Neutron Velocity† [cm/sec]

5. Two-Group Collapsed Assembly Averaged Cross Sections†

   D [cm], Σ_a[cm^{-1}], Σ_r[cm^{-1}],

   $\nu\Sigma_f$[cm^{-1}], $\kappa\Sigma_f$[watt/cm], Σ_f[cm^{-1}]

* These are editing points and do not necessarily correspond to the depletion steps.

† Thermal breakpoint assumed at 0.625 [eV]

FUEL ASSEMBLY STANDARD PROBLEM

II. REACTIVITY DEFECT CALCULATIONS

Provide the following reactivity defects (%Δ k/k) for an assembly with and without burnable poison rods at BOL and EOL (30,000 Mwd/MT):

REACTIVITY DEFECT (%Δ k/k)*	UNPERTURBED CASE†	PERTURBED CASE
1. Fuel Temperature (T_{fuel})	T^{base}_{fuel}	$T^{base}_{moderator}$
2. Moderator Temperature ($T_{moderator}$)	$T^{base}_{moderator}$	$T^{base}_{moderator}$ -25°K
3. Moderator & Fuel Temperature†† ($T_{Moderator}$ & T_{Fuel})	$T^{base}_{moderator}$	68°F
	T^{base}_{fuel}	68°F
4. Moderator & Fuel Temperature†† ($T_{Moderator}$ & T_{Fuel})	$T^{base}_{moderator}$	300°F
	base fuel	300°F
5. Boron Concentration (N_{boron})	N^{base}_{boron}	0 ppm
6. Xenon Concentration (N_{xenon})	Equilibrium	0
7. Control Rod #	Unrodded	Rodded

* It is recommended that a full flux solution be carried out for each state-point.

† Unperturbed parameters are at their base values indicated in the Standard Problem definition.

\# In the case of the W (17x17) assembly only the unpoisoned assembly is required.

†† Pressure is to be maintained at base value.

DATA FOR FUEL ASSEMBLY STANDARD PROBLEM

17 x 17 W Type Fuel Assembly

1. General Characteristics

Power density (W/Gm-U)	38.4
Average fuel temperature (°K)	968
Average clad temperature (°K)	600
Moderator temperature (°K)	560
Soluble boron concentration (ppm)	400
Average core pressure (psia)	2250
Xenon concentration	Equilibrium
Samarium concentration	Equilibrium

2. Configuration (1/8 assembly)

```
4
1 1
1 1 1
2 1 1 3
1 1 1 1 1
1 1 1 1 1 2
3 1 1 2 1 1 1
1 1 1 1 1 1 1 1
1 1 1 1 1 1 1 1 1
```

1 - Fuel Rod
2 - Burnable Poison Rod (BPR)
3 - Guide Thimble
4 - Instrument Thimble

Note: 1. For an unrodded or unpoisoned case replace all BPRs (2) with guide thimbles (3).

2. For a rodded case replace all BPRs (2) with control rods inserted in guide thimbles (3).

3. Fuel Assembly Data

Rod array	17 x 17
Fuel rods per assembly	264
Rod pitch (in)#	0.496
Assembly pitch (in)**	8.466 x 8.466
Assembly length (in)	151.0
Active fuel length (in)	144.0
Number of spacer grids†	8
Compositon of spacer grid	Inconel 718
Weight of spacer grids (lb)	12
Number of guide thimbles	24
Number of instrument thimbles	1

All dimensions are given at cold (68°F) conditions.
† Seven in active length.
** Center to center assembly pitch.

4. Fuel Rod Data

Clad O.D. (in)	0.374
Clad thickness (in)	0.0225
Diametral gap (in)	0.0065
Clad material	Zircaloy-4

5. Fuel Pellet Data

Material	UO_2 - Undished
Density (% of theoretical)	95
Enrichment (w/o)	2.6
Diameter (in)	0.3225

6. Burnable Poison Rod Data (See Figure 1)

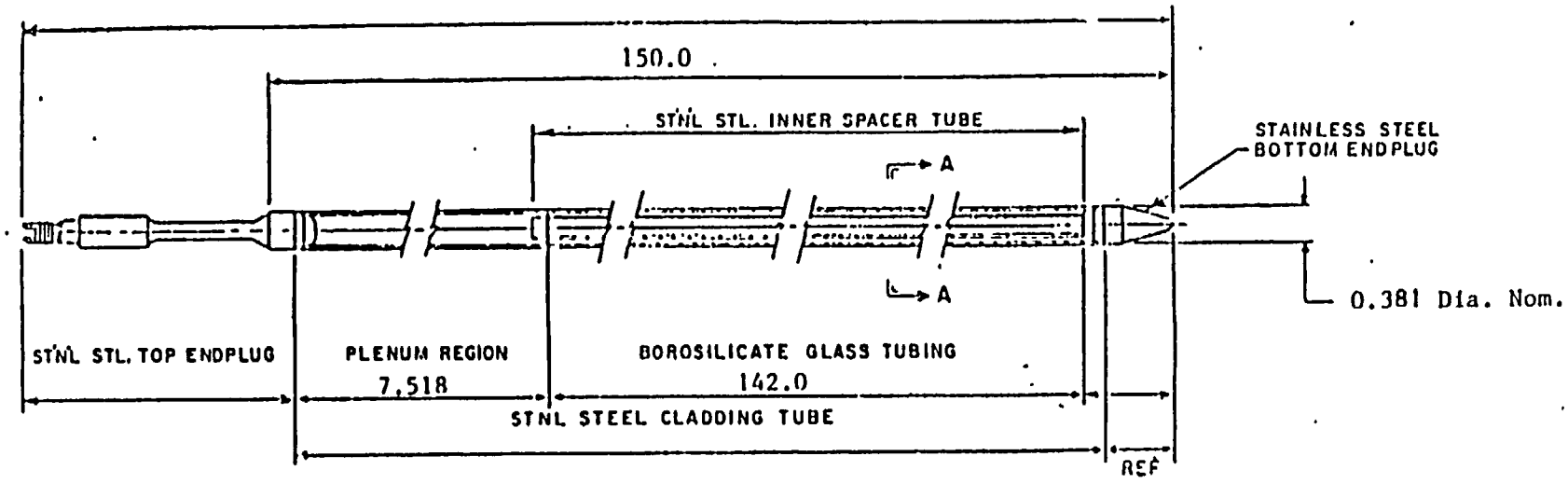
Number per assembly	16
Material	Borosilicate Glass
Density (Borosilicate glass) (gm/cm^3)	2.28
Outside clad O.D. (in)	0.381
Outside Clad I.D. (in)	0.348
Absorber O.D. (in)	0.344
Absorber I.D. (in)	0.185
Inner-tube O.D. (in)	0.1805
Inner-tube I.D. (in)	0.170
Clad material	Stainless Steel
Inner-tube material	Stainless Steel
Boron loading (w/o B_2O_3 in glass rod)	12.5
Weight of Boron-10 (lb/ft)	0.000419

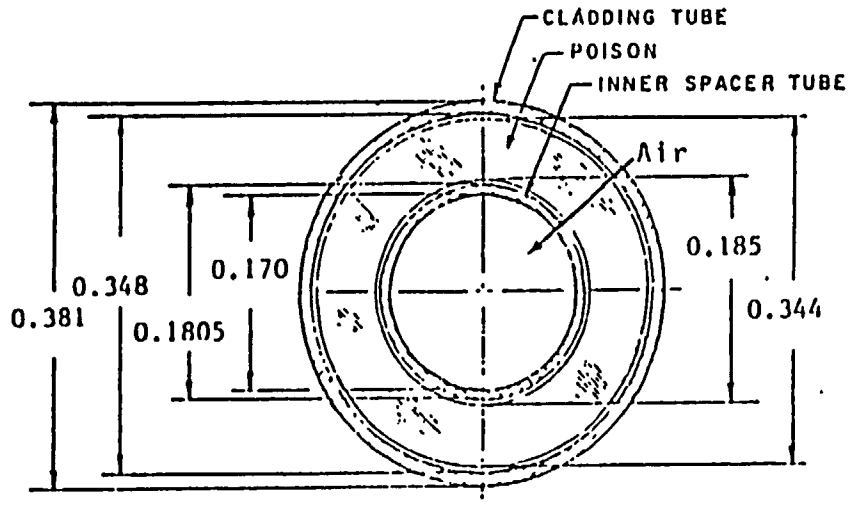
7. Guide Thimbles and Instrument Thimble Data

Number of guide thimbles	24
Number of instrument thimbles	1
Composition of thimbles	Zircaloy-4
Guide Thimble O.D. (in)	0.482
Guide Thimble I.D. (in)	0.450
Instrument Thimble O.D. (in)	0.482
Instrument Thimble I.D. (in)	0.450

8. Control Rod Data

Neutron absorber (w/o)	5% Cd, 15% In, 80% Ag
Absorber diameter (in)	0.341
Absorber density (lb/in^3)	0.367
Cladding material	304 Stainless Steel
Clad O.D. (in)	0.381
Clad thickness (in)	0.0185
Number of control rods	24





SECTION A-A
ENLARGED DETAIL

Figure 1. Burnable Poison Rod Configuration

DESCRIPTION OF CALCULATIONS AND METHODS

1. Name of code/code source/version
2. Reference for calculational method
3. Assembly solution method (Diffusion Theory, Collision Probability, Integral Transport, Monte Carlo, etc.)
4. Pin-cell solution method (if distinct from assembly solution method)
5. Spatial mesh assembly/pin-cell (nxm)
6. Neutron cross sections (ENDF/B or other identification)
7. Number of fast/thermal groups in assembly/pin-cell solution
8. Depletion steps

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NUCLEAR PRODUCTION

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December 19, 1984

Mr. Harold R. Denton, Director
Office of Nuclear Reactor Regulation
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

Attention: Ms. E. G. Adensam, Chief
Licensing Branch No. 4

Subject: McGuire Nuclear Station -- Docket No. 50-369/370
Catawba Nuclear Station - Docket No. 50-413/414
Response Request for Additional Information Regarding
Topical Report DPC-NF-2010, "Nuclear Physics Metodology
for Reload Design"

In response to the request by telephone conference (between NRC, Duke and Brookhaven) on December 17, 1984 for additional information regarding the subject topical report, attached is Duke Power Company's revised answer to question number five, regarding pin power uncertainties.

If any additional information or discussion is desired, please feel free to call Scott Gewehr, Duke Power Licensing at (704) 373-7581.

Very truly yours,

Hal B. Tucker

Hal B. Tucker

SAG/mjf

Attachment

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November 30, 1984

Mr. Harold R. Denton, Director
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U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

Attention: Ms. E. G. Adensam, Chief
Licensing Branch No. 4

Subject: McGuire Nuclear Station - Docket No. 50-369/370
Catawba Nuclear Station - Docket No. 50-413/414
Response Request for Additional Information Regarding
Topical Report DPC-NF-2010, "Nuclear Physics Metodology
for Reload Design"

In response to your request (Reference Letter, E. G. Adensam to H. B. Tucker, November 5, 1984) for additional information regarding the subject topical report, attached are Duke Power Company's answers to the six questions in the request.

If any additional information or discussion is desired, please feel free to call Scott Gewehr, Duke Power Licensing at (704) 373-7581.

Very truly yours,

H. B. Tucker

Hal B. Tucker

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Q.1 Please provide additional information regarding the NUC-MARGINS code and its use in the Dropped Rod Analysis. Provide short descriptions of the input, output, calculational models used, benchmark calculations performed and the conservatisms assumed in the analysis.

A.1 Under the terms of the current fuel contract with Westinghouse, Duke Power will provide physics data for the rod drop transient to Westinghouse who will then perform the safety evaluation and/or reanalysis. This relationship will exist until Duke submits its thermal-hydraulic and safety analysis methodology reports to the NRC.

The physics methods described in Section 4.2.2.5, 6.2.2.4, and 9.1.3.3 will be further elaborated herein.

A. Initial conditions for analysis:

1. Control Bank D is inserted to the Rod Insertion Limit.
2. Core Power is 102% Full Power (2% calorimetric error included).
3. A full power xenon distribution is used which would produce a DNB limiting axial power profile.

B. Assumptions for system response upon rod drop:

1. No trip occurs.
2. Control bank D is withdrawn to compensate for the dropped rod.
3. A short duration reactor power overshoot will occur with the turbine-reactor control system eventually leveling out the reactor power to the initial power level.

Search cases are performed as described in Section 4.2.2.5 and 6.2.2.4. EPRI-NODE assembly average powers are converted to $F^N_{\Delta H}$ using the method described below. This method is employed for all $F^N_{\Delta H}$ evaluations. All physics codes employed are static, therefore, "before" and "after" rod drop power distributions are calculated.

The mathetical formulation of $F^C_{\Delta H}$ employs the Section 6.2.1.2 definitions as follows:

$$F^C_{\Delta H,j} = [\sum_{i=1}^{K} F^{node}_{i,j} \times RL^N_j + FR_j \times F^{node}_{K,j} \times RL^N_j + (1-FR_j) \times F^{node}_{K,J} \times RL^R_j + \sum_{i=N+1}^{M} F^{node}_{i,j} \times RL^R_j] \div M$$

and then:

$$F^C_{\Delta H} = \underset{j}{Max} \, (F^C_{\Delta H,j})$$

Where:

M = Number of axial nodes.

RL_j^N = Non-rodded radial local factor for assembly j.

RL^R_j = Rodded radial local factor for assembly j.

FR_j = Linear fraction of assembly j which does not contain a control rod.

Radial local factors are edited by PDQ-EDIT using fine mesh PDQØ7 mesh average powers. The PDOØ7 cases are two-dimensional simulations with control bank(s) explicitly represented.

The nodal powers, $F_{i,j}^{node}$, are steady state three-dimensional calculations which explicitly model; control bank insertion, boron and xenon conditions, and other reactor state point variables necessary for a best estimate power distribution calculation.

$F_{\Delta H}^C$ is then evaluated by the NUC-MARGINS code or by hand calculations using the nodal powers from NODE-P and the RL from PDQO7. The NUC-MARGINS code has been independently verified to yield the correct $F_{\Delta H}^N$. $F_{\Delta H}^T$ is the ultimate output as defined by equation 6-2 for DNB analysis.

The system transient response and the transient DNB calculations would be performed by Westinghouse if the physics parameters exceeded the bounds of the previous analyses.

Q.2 Identify the nominal and various off-nominal cross-section sets that are generated in order to evaluate the different reactivity coefficients and defects.

A.2 The various fuel cross-section sets that are generated in order to evaluate different reactivity coefficients and defects are identified in Table 2.1. Nominal cross-sections are generated as a function of burnup at an average moderator temperature of 594°F and an average fuel temperature of 1250°F. The off-nominal cross-sections are generated at various burnups with varying moderator and fuel temperatures.

The cross-section representation in PDQØ7 differs between the quarter-core discrete pin and colorset models. The representation employed in the quarter-core model is discussed first and then the colorset discussion follows. All sets, except the baffle, use combined macroscopic and microscopic cross-sections.

Fuel cross-sections in quarter-core PDQØ7 are calculated according to the following relation:

$$\Sigma(T_M, T_F, Bu) = \Sigma_o(Bu) + \frac{\Delta\Sigma}{\Delta T_M} \times (T_M - T_M^{Ref}) + \frac{\Delta\Sigma}{\Delta\sqrt{T_F}} \times (\sqrt{T_F} - \sqrt{T_F}^{Ref})$$

where $\Sigma(T_M, T_F, Bu)$ = the total macroscopic cross-section as a function of moderator temperature, fuel temperature, and burnup.

$\Sigma_o(Bu)$ = the nominal macroscopic cross section as a function of burnup.

$\frac{\Delta\Sigma}{\Delta T_M}$ = the moderator temperature pseudo-microscopic cross-section which relates the change in macroscopic cross-section to change in moderator temperature.

$\frac{\Delta\Sigma}{\Delta\sqrt{T_F}}$ = the fuel temperature pseudo-microscopic cross-section which relates the change in macroscopic cross-section to a change in fuel temperature.

The macroscopic cross-sections given here may be of any type, e.g. transport, absorption, removal, or fission. The pseudo-microscopic cross-sections (or pseudo-micros) account for the change in the macroscopic cross-section as a result of a change from reference conditions. These pseudo-micros are input to PDQØ7 as a function of burnup. The moderator temperature pseudo-micros are determined from the cross-section sets at moderator temperatures of 630°F and 530°F (fuel temperature held constant at 1250°F).

The fuel temperature pseudo-micros are determined from the cross-section sets at fuel temperatures of 1250°F and 594°F (moderator temperature held constant at 594°F).

Most nonfuel cross-sections employed in quarter core calculations are evaluated as shown in Table 2.4, and are consistent with the core average moderator temperature of interest.

The reflector constants are evaluated at T_{inlet} (usually 557°F) and, at Hot Zero Power, are identical to the water gap constants. Baffle constants are evaluated using the method shown in Chapter 4 of EPRI NP-3642-SR (Few-Group Baffle and/or Reflector Constants for Diffusion Calculation Application, EPRI Special Report, August 1984).

Colorset PDQØ7 calculations are performed which provide sufficient data to characterize operation from Hot Full Power (HFP) to Cold Zero Power (CZP) conditions. A breakpoint is designated at Hot Zero Power (HZP). Two sets of data (B-Constants) are then used in EPRI-NODE-P calculations:

1. Normal Operation - HFP to HZP

2. Low Temperature - HZP to CZP

B-Constants for the Normal Operation and Low Temperature models are generated following the sequence described in Section 3 of DPC-NF-2010.

Tables 2.1 and 2.4 describe conditions for fuel and non-fuel cross-section sets. The Normal Operation cross-sections input to colorset PDQØ7 calculations are shown by the matrices in Table 2.2. Table 2.3 shows matrices of cross section sets for Low Temperature colorset calculations. Nonfuel cross-section sets (Table 2.4) are used which are consistent with the fuel moderator temperature.

Table 2.1

McGuire/Catawba
Fuel Cross-Section Sets

Cross-Section Set Type	T_{mod} (°F)	T_{fuel} (°F)	Power	Burnup Timesteps (GWD/MTU)	Application
P1	594	594	Zero	0.0	HFP → HZP
P2 (Nominal)	594	1250	Full	0.0	"
P3	630	1250	Full	0.0	"
P4	530	1250	Full	0.0	"
P8 (Nominal)	594	1250	Full	0.0, 0.1, 0.5, 1.0, 2.0, 4.0, 6.0, ..., 58.0, 60.0	"
P8B6	594	594	Full	"	"
P8B7	530	1250	Full	"	"
P8B8	630	1250	Full	"	"
P5	200	200	Zero	0.0	HZP → CZP
P9	200	200	Zero	0.0, 0.1, 0.5, 1.0, 2.0, 4.0, 6.0, ..., 58.0, 60.0	"
P6	557	557	Zero	0.0	"
P7	68	68	Zero	0.0	"

Table 2.2

Cross-Section Sets for Normal Operation
PDQØ7 Colorsets

BOL

Effect	Cross-Section Set Type			
	P2(Nominal)	P1	P3	P4
Soluble Boron	X			
K-inf vs. T_{mod}	X		X	X
Migration Area vs. T_{mod}	X		X	X
Doppler	X	X		

Depletion

Reactivity Effect	Cross-Section Set Type			
	P8(Nominal)	P8B6	P8B7	P8B8
Exposure	X			
Soluble Boron	X			
Control Rods	X			
Xenon	X			
Doppler	X	X		
Moderator			X	X

Table 2.3

Cross-Section Sets for Low Temperature
PDQØ7 Colorsets

BOL

Effect	Cross-Section Set Type P5	P6	P7
Soluble Boron	X		
K-inf. vs. T_{mod}	X	X	X
Migration Area vs. T_{mod}	X	X	X

DEPLETION

Reactivity Effect	Cross-Section Set Type P9
Exposure	X
Soluble Boron	X
Control Rods	X

Table 2.4

McGuire/Catawba
Non-fuel Cross-Section Sets

Material	Moderator Temperatures (°F)
Water Gap/Reflector	630, 594, 557, 530, 200, 68
Guide Tube/Inst. Tube	630, 594, 557, 530, 200, 68
Control Rod	594, 557, 200, 68
Burnable Poison Rod	594, 557, 200, 68
Baffle	EPRI NP-3642-SR

Q.3 Provide a short description of the PDQ-EDIT code and describe the verification program that was undertaken to test data generated with PDQ-EDIT for use in SNA-CORE.

A.3 PDQ-EDIT is a utility code written by Duke Power Company that is capable of reading Internal File Management (IFM) files written by PDQ07. This code is primarily used to develop theoretical factors for SNA-CORE, and to edit and process data contained on pointwise flux, power and concentration IFM files. PDQ-EDIT, like all Nuclear Design software used in safety related analysis, is quality assured as required by Duke Power Company's Administrative Policy Manual for Nuclear Stations.

SNA-CORE theoretical factors are generated from PDQ-EDIT in what is commonly known as theoretical factor sets. Each theoretical factor set is valid over a user defined burnup range. Theoretical factor sets consist of assembly average powers, assembly peak pin powers, and detector mesh average two-group fluxes.

Verification of theoretical factor sets is accomplished by the utility code SNAVER. SNAVER compares the symmetric assembly average and peak pin powers on either a 1/4-core or 1/8-core basis, and then calculates a percent difference for each power at a given location with respect to the average at that location. Percent differences greater than 0.1% are flagged by the program. The cognizant engineer must then verify whether these errors are justified. SNAVER also checks for consistancy between detector fluxes at symmetric locations, and for correct data format.

The formal benchmarking of theoretical factors developed from PDQ-EDIT was accomplished by comparing measured powers from Westinghouse's INCORE code, to those calculated from SNA-CORE for Sequoyah Unit 1 Cycle 1. All measured powers were inferred from plant supplied flux traces. Results from these comparisons are shown in Figures 1 thru 7. Good agreement between the two codes was observed. A summary of the average absolute relative error, and the standard deviation associated with these errors are presented in Table 1.

In conclusion, comparisons between measured data from Westinghouse's INCORE code and Duke's SNA-CORE code demonstrate the accuracy of the PDQ07, PDQ-EDIT, SNA-CORE code package. Also, in addition to the software quality assurance program employed at Duke, SNAVER provides an independent means of verifying the correctness of theoretical factor sets before they are used in a production environment.

Table 1

Statistical Summary of INCORE versus SNA-CORE
Measured Powers for Sequoyah 1 Cycle 1

CASE	Burnup EFPD	Average Absolute Relative Error (%)	Standard Derration %
1	71.82	1.34	1.84
2	101.62	1.06	1.43
3	133.30	1.14	1.48
4	166.04	1.28	1.64
5	231.70	1.21	1.48
6	292.04	1.20	1.51
7	378.92	1.05	1.34

Average Absolute Relative Error $(\bar{D}) \equiv \left|[(\text{SNA-CORE} - \text{INCORE})/\text{INCORE}]\right| * 100$

$$\bar{D} \equiv \sum_{i=1}^{N} D_i/N$$

FIGURE 1

SEQUOYAH 1 CYCLE 1 SWA-CORE VS. INCORE MEASURED POWERS
71.82 EFPD 100(%)FP CONTROL BANK D AT 200 STEPS WITHDRAWN

	H	G	F	E	D	C	B	A	
	1.12	1.05	1.17	1.11	1.15	1.05	1.01	.71	
8	1.17	1.08	1.17	1.14	1.18	1.07	1.01	.71	
		1.16	1.11	1.19	1.16	1.13	1.01	.77	
9		1.17	1.13	1.19	1.17	1.13	1.03	.77	
			1.18	1.12	1.18	1.09	.98	.66	
10			1.17	1.14	1.18	1.11	.97	.65	
				1.18	1.13	1.08	.92	.56	
11				1.19	1.16	1.08	.92	.55	
					1.09	.99	.86		
12					1.12	.99	.83		
						1.02	.51		SWA-CORE
13						.99	.49		INCORE

FIGURE 2

SEQUOYAH 1 CYCLE 1 SNA-CORE VS. INCORE MEASURED POWERS
101.62 EFPD 100(%)FP CONTROL BANK D AT 218 STEPS WITHDRAWN

	H	G	F	E	D	C	B	A	
	1.14	1.06	1.16	1.13	1.17	1.06	1.00	.71	
8	1.16	1.09	1.17	1.15	1.17	1.08	1.00	.71	
		1.16	1.12	1.18	1.16	1.12	1.01	.76	
9		1.17	1.14	1.18	1.18	1.13	1.03	.76	
			1.18	1.13	1.17	1.09	.97	.65	
10			1.17	1.15	1.17	1.11	.96	.65	
				1.17	1.13	1.08	.91	.55	
11				1.18	1.16	1.08	.92	.55	
					1.11	1.00	.85		
12					1.11	1.00	.83		
						1.02	.51	SNA-CORE	
13						.99	.50	INCORE	

FIGURE 3

SEQUOYAH 1 CYCLE 1 SNA-CORE VS. INCORE MEASURED POWERS
133.30 EFPD 100(%)FP CONTROL BANK D AT 216 STEPS WITHDRAWN

	H	G	F	E	D	C	B	A	
	1.14	1.08	1.17	1.14	1.16	1.07	.99	.70	
8	1.16	1.11	1.17	1.17	1.17	1.09	.99	.71	
		1.17	1.14	1.19	1.17	1.12	1.01	.76	
9		1.17	1.16	1.18	1.19	1.12	1.03	.76	
			1.18	1.14	1.17	1.09	.96	.65	
10			1.17	1.16	1.16	1.11	.95	.65	
				1.16	1.13	1.06	.91	.55	
11				1.17	1.16	1.06	.92	.55	
					1.09	1.00	.84		
12					1.10	1.00	.82		
						1.01	.51		SNA-CORE
13						.98	.50		INCORE

FIGURE 4

SEQUOYAH 1 CYCLE 1 SNA-CORE VS. INCORE MEASURED POWERS
166.04 EFPD 100(%)FP CONTROL BANK D AT 210 STEPS WITHDRAWN

	H	G	F	E	D	C	B	A	
	1.13	1.09	1.17	1.15	1.14	1.08	.99	.71	
8	1.16	1.11	1.17	1.18	1.15	1.10	.99	.71	
		1.16	1.15	1.19	1.17	1.11	1.01	.76	
	9	1.17	1.18	1.18	1.19	1.11	1.03	.76	
			1.18	1.15	1.16	1.09	.96	.66	
		10	1.17	1.18	1.15	1.11	.95	.65	
				1.16	1.13	1.06	.91	.55	
			11	1.17	1.17	1.06	.92	.55	
					1.08	1.00	.84		
				12	1.07	1.00	.82		
						1.00	.51		SNA-CORE
					13	.97	.50		INCORE

FIGURE 5

SEQUOYAH 1 CYCLE 1 SNA-CORE VS. INCORE MEASURED POWERS
231.70 EFPD 100(%)FP CONTROL BANK D AT 216 STEPS WITHDRAWN

	H	G	F	E	D	C	B	A	
	1.10	1.08	1.14	1.16	1.13	1.09	.99	.72	
8	1.12	1.10	1.14	1.19	1.13	1.12	.99	.73	
		1.13	1.16	1.16	1.17	1.09	1.02	.76	
9		1.14	1.18	1.15	1.19	1.09	1.04	.77	
			1.16	1.16	1.14	1.10	.96	.67	
10			1.15	1.18	1.13	1.12	.95	.68	
				1.14	1.14	1.05	.92	.56	
11				1.14	1.17	1.05	.93	.56	
					1.07	1.02	.84		
12					1.08	1.02	.82		
						1.00	.53		SNA-CORE
13						.98	.52		INCORE

FIGURE 6

SEQUOYAH 1 CYCLE 1 SNA-CORE VS. INCORE MEASURED POWERS
292.04 EFPD 100(%)FP CONTROL BANK D AT 216 STEPS WITHDRAWN

	H	G	F	E	D	C	B	A	
	1.07	1.07	1.12	1.15	1.11	1.10	.99	.74	
8	1.09	1.09	1.12	1.18	1.12	1.13	.99	.75	
		1.11	1.15	1.14	1.16	1.08	1.03	.78	
9		1.11	1.18	1.13	1.18	1.08	1.05	.78	
			1.14	1.16	1.12	1.11	.97	.69	
10			1.13	1.18	1.11	1.13	.96	.69	
				1.12	1.13	1.05	.93	.58	
11				1.12	1.16	1.05	.94	.57	
					1.06	1.04	.85		
12					1.07	1.04	.83		
						1.00	.55		SNA-CORE
13						.98	.54		INCORE

FIGURE 7

SEQUOYAH 1 CYCLE 1 SNA-CORE VS. INCORE MEASURED POWERS
378.92 EFPD 100(%)FP CONTROL BANK D AT 222 STEPS WITHDRAWN

	H	G	F	E	D	C	B	A	
	1.02	1.04	1.08	1.14	1.09	1.10	1.00	.77	
8	1.06	1.06	1.08	1.15	1.09	1.13	1.01	.79	
		1.07	1.13	1.10	1.14	1.07	1.05	.80	
9		1.08	1.15	1.10	1.15	1.07	1.07	.81	
			1.10	1.14	1.09	1.11	.98	.73	
10			1.09	1.15	1.09	1.13	.97	.74	
				1.09	1.13	1.05	.96	.60	
11				1.10	1.15	1.05	.97	.60	
					1.06	1.06	.87		
12					1.07	1.06	.85		
						1.02	.58		SNA-CORE
13						1.00	.57		INCORE

Q.4. Comment on the reasons for the 3.1% non-conservative bias in the calculated peak axial powers (Section 11.5.4). Describe the model refinements, if any, that have been undertaken to reduce this bias.

A.4. The reason there is a -0.031 bias on the calculated peak axial powers (Section 11.5.4) is that the models used by Duke at the time of this report underpredicted the peak axial power. This -0.031 bias is the mean difference ($\overline{D}$) and is defined by equation 11-2. This value is a difference and not a percentage difference. The mean percent difference for all cases considered was -2.195% (Table 11-10). Again, it should be pointed out, that this number applies to all peak C, M pairs $\geq$ 1.0.

Although Dukes' models underpredict the peak axial power on an average of -2.195%, the Observed Nuclear Reliability Factor (ONRF) directly reflects this non-conservative prediction. This can be seen by examining equation 11-11. Because $\overline{D}$ is subtracted from $\overline{M}$, this equation is conservative for all cases of $\overline{D}$. (That is, $\overline{D}$ being positive, negative, or 0)

Consider the ONRF calculation of the peak axial power on Table 11-6. In this example if $\overline{D}$ were 0 the ONRF would be 1.035. With a $\overline{D}$ of -0.031 the ONRF is 1.058. This is a 2.2% increase in ONRF. The $\overline{D}$ of -0.031 represents a 2.195% underprediction of measured peak axial power. (Table 11-10). Therefore, it can be seen from this example, that there is a 1% increase in ONRF for each 1% that the model underpredicts the measured peak axial power.

In summary, even though the models used by Duke underpredict the peak axial power, the ONRF reflects this underprediction. As shown in the above example, there is a 1 to 1 correspondence in the percentage of the underprediction to the percentage increase in the ONRF.

The model refinements undertaken to reduce this underprediction are discussed in the answer to question 6 parts one and two. The refinements are; 1) normalization of EPRI-NODE-P to include unrodded M^2 adjustments, and 2) an increase in the number of axial nodes. Attached are the results of some maps compared to predictions using 12 levels and 18 levels of EPRI-NODE-P. Attached are the Difference Means and Standard Deviations for Assembly Peak Axial Powers (C, M $\geq$ 1.0), and Assembly Radial Powers. Also attached are Percent Difference Means (C, M $\geq$ 1.0) for Assembly Peak Axial Powers and Assembly Radial Powers.

Table 4-1

Difference Means and Standard Deviations for Assembly Radial Powers
(C, M $\geq$ 1.0)

Unit/Cycle	EPRI-NODE-P Model	N	$\overline{D}$	$S(\overline{D})$	$\overline{ABS(D)}$	S(ABS(D))
M1/C2	12 Level	144	-0.002	0.017	0.014	0.010
M1/C2	18 Level	144	-0.002	0.015	0.012	0.010

Difference Means and Standard Deviations for Assembly Peak Axial Powers
(C, M $\geq$ 1.0)

Unit/Cycle	EPRI-NODE-P Model	N	$\overline{D}$	$S(\overline{D})$	$\overline{ABS(D)}$	S(ABS(D))
M1/C2	12 Level	232	-0.004	0.031	0.025	0.018
M1/C2	18 Level	246	0.030	0.035	0.036	0.029

Percent Difference Means for Assembly Radial Powers
(C, M $\geq$ 1.0)

Unit/Cycle	EPRI-NODE-P Model	Mean % Difference	Mean Absolute % Difference
M1/C2	12 Level	-0.170	1.35
M1/C2	18 Level	-0.142	1.17

Percent Difference Means for Assembly Peak Axial Powers
(C, M $\geq$ 1.0)

Unit/Cycle	EPRI-NODE-P Model	Mean % Difference	Mean Absolute % Difference
M1/C2	12 Level	-0.407	2.039
M1/C2	18 Level	2.382	2.890

FIGURE 4.1

MCGUIRE-1 CYCLE-2 ASSEMBLY PEAK AXIAL POWERS - CALC (12 LEVEL) VS. MEAS

18 EFPD 100%FP CONTROL BANK D AT 207 STEPS WITHDRAWN

	H	G	F	E	D	C	B	A
8	.95 .93	1.08 1.06	1.24 1.27	.97 .98	.93 1.00	.80 .5	1.09 1.19	1.28 1.27
9	1.10 1.09	1.27 1.27	1.25 1.25	1.03 1.03	.98 1.02	.93 .95	1.50 1.53	1.30 1.28
10	1.24 1.28	1.25 1.27	1.25 1.27	1.28 1.32	1.00 1.03	.96 1.00	1.13 1.19	1.19 1.16
11	.98 1.00	1.04 1.04	1.28 1.32	1.25 1.28	1.27 1.29	1.14 1.15	1.52 1.48	.92 .91
12	.94 1.02	.99 1.04	1.01 1.02	1.27 1.30	1.43 1.40	1.43 1.41	1.29 1.26	
13	.81 .88	.93 .98	.97 1.05	1.14 1.17	1.43 1.44	.99 .98	.79 .77	
14	1.10 1.12	1.51 1.46	1.14 1.14	1.52 1.44	1.30 1.26	.80 .79		
15	1.28 1.27	1.31 1.26	1.19 1.15	.93 .90	CALC MEAS			

FIGURE 4.2

MCGUIRE-1 CYCLE-2 ASSEMBLY PEAK AXIAL POWERS - CALC (12 LEVEL) VS. MEAS

30 EFPD 100%FP CONTROL BANK D AT 194 STEPS WITHDRAWN

	H	G	F	E	D	C	B	A
	.90	1.04	1.21	.95	.92	.82	1.12	1.30
8	.92	1.06	1.26	.98	1.02	.89	1.20	1.30
	1.05	1.24	1.22	1.00	.98	.95	1.53	1.33
9	1.08	1.26	1.25	1.03	1.04	1.00	1.53	1.29
	1.22	1.23	1.24	1.30	1.01	.98	1.16	1.21
10	1.27	1.26	1.26	1.32	1.04	1.04	1.21	1.19
	.95	1.01	1.30	1.26	1.28	1.15	1.54	.94
11	1.00	1.04	1.33	1.29	1.31	1.17	1.52	.94
	.92	.98	1.02	1.28	1.43	1.43	1.31	
12	1.03	1.05	1.04	1.31	1.42	1.44	1.29	
	.83	.96	.98	1.16	1.44	1.00	.80	
13	.90	1.01	1.08	1.19	1.46	1.01	.79	
	1.12	1.54	1.16	1.55	1.31	.81		
14	1.15	1.50	1.16	1.45	1.28	.81		
	1.31	1.33	1.21	.94	CALC			
15	1.30	1.30	1.16	.90	MEAS			

FIGURE 4.3

MCGUIRE-1 CYCLE-2 -ASSEMBLY PEAK AXIAL POWERS - CALC (12 LEVEL) VS. MEAS

48 EFPD 100%FP CONTROL BANK D AT 228 STEPS WITHDRAWN

	H	G	F	E	D	C	B	A
8	.94 .92	1.08 1.04	1.22 1.23	.98 .97	.95 1.00	.81 .86	1.07 1.16	1.24 1.25
9	1.10 1.07	1.25 1.24	1.23 1.22	1.04 1.01	1.00 1.01	.92 .97	1.46 1.48	1.27 1.24
10	1.23 1.25	1.24 1.23	1.24 1.22	1.26 1.27	1.01 1.00	.94 1.01	1.11 1.17	1.15 1.14
11	.99 .99	1.04 1.03	1.26 1.26	1.22 1.22	1.22 1.25	1.10 1.12	1.47 1.45	.90 .90
12	.96 1.00	1.01 1.02	1.01 1.00	1.22 1.25	1.35 1.34	1.36 1.37	1.24 1.23	
13	.82 .87	.92 .97	.94 1.04	1.10 1.14	1.36 1.39	.95 .96	.77 .76	
14	1.08 1.11	1.47 1.45	1.11 1.12	1.47 1.39	1.24 1.23	.78 .78		
15	1.24 1.26	1.27 1.25	1.16 1.12	.90 .87	CALC MEAS			

FIGURE 4.4

MCGUIRE-1 CYCLE-2 ASSEMBLY PEAK AXIAL POWERS - CALC (12 LEVEL) VS. MEAS

61 EFPD 100%FP CONTROL BANK D AT 220 STEPS WITHDRAWN

	H	G	F	E	D	C	B	A
	.92	1.05	1.19	.97	.94	.81	1.08	1.25
8	.91	1.03	1.23	.96	1.00	.86	1.15	1.24
	1.07	1.22	1.20	1.02	.99	.93	1.47	1.27
9	1.06	1.23	1.21	1.00	1.00	.96	1.47	1.24
	1.20	1.21	1.21	1.24	1.00	.95	1.12	1.16
10	1.24	1.22	1.21	1.26	1.00	1.01	1.17	1.14
	.97	1.02	1.24	1.20	1.22	1.11	1.47	.91
11	.99	1.03	1.26	1.22	1.24	1.12	1.45	.90
	.95	1.00	1.00	1.22	1.35	1.36	1.24	
12	1.01	1.02	1.00	1.24	1.33	1.35	1.22	
	.82	.93	.95	1.11	1.36	.95	.77	
13	.88	.98	1.04	1.14	1.38	.96	.75	
	1.09	1.48	1.12	1.48	1.24	.78		
14	1.10	1.44	1.12	1.38	1.22	.77		
	1.25	1.27	1.16	.91	CALC			
15	1.24	1.23	1.11	.87	MEAS			

FIGURE 4.5

MCGUIRE-1 CYCLE-2 ASSEMBLY PEAK AXIAL POWERS - CALC (12 LEVEL) VS. MEAS

101 EFPD 100%FP CONTROL BANK D AT 223 STEPS WITHDRAWN

	H	G	F	E	D	C	B	A
8	.91 .90	1.03 1.03	1.16 1.23	.96 .97	.95 1.01	.83 .88	1.08 1.14	1.22 1.21
9	1.04 1.04	1.19 1.21	1.17 1.19	1.00 1.00	.99 1.01	.94 .97	1.46 1.45	1.25 1.21
10	1.17 1.22	1.17 1.20	1.17 1.19	1.23 1.25	.99 1.00	.96 1.01	1.11 1.15	1.14 1.13
11	.96 .99	1.01 1.02	1.23 1.25	1.19 1.21	1.20 1.23	1.09 1.10	1.45 1.41	.90 .90
12	.95 1.01	1.00 1.02	.99 1.00	1.20 1.23	1.31 1.30	1.32 1.32	1.21 1.20	
13	.84 .89	.94 .98	.96 1.03	1.09 1.12	1.32 1.34	.94 .95	.77 .76	
14	1.09 1.10	1.46 1.43	1.11 1.11	1.45 1.37	1.21 1.20	.77 .77		
15	1.22 1.21	1.25 1.20	1.14 1.10	.90 .87	CALC MEAS			

FIGURE 4.6

MCGUIRE-1 CYCLE-2 ASSEMBLY PEAK AXIAL POWERS - CALC (12 LEVEL) VS. MEAS

130 EFPD 100%FP CONTROL BANK D AT 216 STEPS WITHDRAWN

	H	G	F	E	D	C	B	A
8	.90 .93	1.02 1.04	1.18 1.23	.95 .98	.95 1.01	.85 .89	1.09 1.14	1.21 1.20
9	1.04 1.07	1.20 1.23	1.18 1.20	1.00 1.01	.99 1.02	.96 .98	1.46 1.45	1.24 1.20
10	1.18 1.23	1.18 1.22	1.18 1.20	1.24 1.26	1.00 1.01	.97 1.02	1.12 1.14	1.14 1.11
11	.95 .99	1.00 1.03	1.24 1.26	1.19 1.22	1.20 1.24	1.10 1.11	1.44 1.41	.90 .90
12	.95 1.03	1.00 1.04	1.00 1.02	1.20 1.24	1.30 1.31	1.31 1.32	1.20 1.19	
13	.85 .90	.96 1.00	.97 1.05	1.10 1.13	1.31 1.34	.95 .95	.77 .76	
14	1.09 1.10	1.46 1.43	1.12 1.11	1.44 1.37	1.20 1.19	.78 .77		
15	1.21 1.19	1.24 1.20	1.14 1.10	.90 .87	CALC MEAS			

FIGURE 4.7

MCGUIRE-1 CYCLE-2 ASSEMBLY PEAK AXIAL POWERS - CALC (18 LEVEL) VS. MEAS

18 EFPD 100%FP CONTROL BANK D AT 207 STEPS WITHDRAWN

	H	G	F	E	D	C	B	A
8	.96 .93	1.11 1.06	1.27 1.27	1.00 .98	.96 1.00	.82 .85	1.13 1.19	1.32 1.27
9	1.13 1.09	1.30 1.27	1.28 1.25	1.06 1.03	1.01 1.02	.95 .95	1.55 1.53	1.35 1.28
10	1.27 1.28	1.28 1.27	1.29 1.27	1.32 1.32	1.03 1.03	.99 1.00	1.17 1.19	1.23 1.16
11	1.00 1.00	1.06 1.04	1.32 1.32	1.29 1.28	1.31 1.29	1.17 1.15	1.57 1.48	.95 .91
12	.97 1.02	1.02 1.04	1.03 1.02	1.31 1.30	1.47 1.40	1.47 1.41	1.33 1.26	
13	.83 .88	.96 .98	.99 1.05	1.17 1.17	1.47 1.44	1.01 .98	.82 .77	
14	1.13 1.12	1.56 1.46	1.17 1.14	1.57 1.44	1.34 1.26	.82 .79		
15	1.33 1.27	1.35 1.26	1.23 1.15	.96 .90	CALC MEAS			

FIGURE 4.8

MCGUIRE-1 CYCLE-2 ASSEMBLY PEAK AXIAL POWERS - CALC (18 LEVEL) VS. MEAS

30 EFPD 100%FP CONTROL BANK D AT 194 STEPS WITHDRAWN

	H	G	F	E	D	C	B	A
8	.92 .92	1.06 1.06	1.24 1.26	.97 .98	.94 1.02	.84 .89	1.15 1.20	1.34 1.30
9	1.08 1.08	1.27 1.26	1.26 1.25	1.03 1.03	1.01 1.04	.98 1.00	1.58 1.53	1.37 1.29
10	1.25 1.27	1.26 1.26	1.27 1.26	1.34 1.32	1.04 1.04	1.01 1.04	1.19 1.21	1.25 1.19
11	.98 1.00	1.04 1.04	1.34 1.33	1.30 1.29	1.32 1.31	1.19 1.17	1.59 1.52	.97 .94
12	.95 1.03	1.01 1.05	1.04 1.04	1.32 1.31	1.47 1.42	1.48 1.44	1.35 1.29	
13	.85 .90	.98 1.01	1.01 1.08	1.19 1.19	1.48 1.46	1.03 1.01	.83 .79	
14	1.16 1.15	1.59 1.50	1.20 1.16	1.59 1.45	1.35 1.28	.84 .81		
15	1.35 1.30	1.38 1.30	1.25 1.16	.97 .90	CALC MEAS			

FIGURE 4.9

MCGUIRE-1 CYCLE-2 ASSEMBLY PEAK AXIAL POWERS - CALC (18 LEVEL) VS. MEAS

48 EFPD 100%FP CONTROL BANK D AT 228 STEPS WITHDRAWN

	H	G	F	E	D	C	B	A
8	.96 .92	1.10 1.04	1.25 1.23	1.00 .97	.98 1.00	.84 .86	1.11 1.16	1.28 1.25
9	1.12 1.07	1.28 1.24	1.26 1.22	1.06 1.01	1.03 1.01	.95 .97	1.51 1.48	1.31 1.24
10	1.26 1.25	1.27 1.23	1.27 1.22	1.30 1.27	1.04 1.00	.97 1.01	1.14 1.17	1.19 1.14
11	1.01 .99	1.07 1.03	1.30 1.26	1.26 1.22	1.26 1.25	1.14 1.12	1.52 1.45	.93 .90
12	.98 1.00	1.03 1.02	1.04 1.00	1.26 1.25	1.39 1.34	1.40 1.37	1.28 1.23	
13	.84 .87	.95 .97	.97 1.04	1.14 1.14	1.40 1.39	.98 .96	.79 .76	
14	1.11 1.11	1.52 1.45	1.15 1.12	1.52 1.39	1.28 1.23	.80 .78		
15	1.29 1.26	1.31 1.25	1.20 1.12	.93 .87	CALC MEAS			

FIGURE 4.10

MCGUIRE-1 CYCLE-2 ASSEMBLY PEAK AXIAL POWERS - CALC (18 LEVEL) VS. MEAS

61 EFPD 100%FP CONTROL BANK D AT 220 STEPS WITHDRAWN

	H	G	F	E	D	C	B	A
	.95	1.08	1.23	.99	.97	.84	1.12	1.29
8	.91	1.03	1.23	.96	1.00	.86	1.15	1.24
	1.10	1.25	1.24	1.05	1.02	.96	1.52	1.31
9	1.06	1.23	1.21	1.00	1.00	.96	1.47	1.24
	1.23	1.24	1.25	1.28	1.03	.98	1.15	1.20
10	1.24	1.22	1.21	1.26	1.00	1.01	1.17	1.14
	1.00	1.05	1.28	1.24	1.26	1.14	1.52	.93
11	.99	1.03	1.26	1.22	1.24	1.12	1.45	.90
	.98	1.02	1.03	1.26	1.39	1.40	1.28	
12	1.01	1.02	1.00	1.24	1.33	1.35	1.22	
	.85	.96	.98	1.14	1.40	.98	.80	
13	.88	.98	1.04	1.14	1.38	.96	.75	
	1.12	1.53	1.16	1.52	1.28	.80		
14	1.10	1.44	1.12	1.38	1.22	.77		
	1.29	1.31	1.20	.94	CALC			
15	1.24	1.23	1.11	.87	MEAS			

FIGURE 4.11

MCGUIRE-1 CYCLE-2 ASSEMBLY PEAK AXIAL POWERS - CALC (18 LEVEL) VS. MEAS

101 EFPD 100%FP CONTROL BANK D AT 223 STEPS WITHDRAWN

	H	G	F	E	D	C	B	A
	.93	1.05	1.20	.98	.97	.85	1.12	1.26
8	.90	1.03	1.23	.97	1.01	.88	1.14	1.21
	1.07	1.23	1.21	1.03	1.02	.97	1.51	1.29
9	1.04	1.21	1.19	1.00	1.01	.97	1.45	1.21
	1.21	1.21	1.21	1.27	1.02	.99	1.15	1.18
10	1.22	1.20	1.19	1.25	1.00	1.01	1.15	1.13
	.98	1.03	1.27	1.23	1.24	1.13	1.49	.93
11	.99	1.02	1.25	1.21	1.23	1.10	1.41	.90
	.98	1.02	1.02	1.24	1.35	1.36	1.25	
12	1.01	1.02	1.00	1.23	1.30	1.32	1.20	
	.86	.98	.99	1.13	1.36	.97	.79	
13	.89	.98	1.03	1.12	1.34	.95	.76	
	1.12	1.51	1.15	1.49	1.25	.80		
14	1.10	1.43	1.11	1.37	1.20	.77		
	1.26	1.29	1.18	.93	CALC			
15	1.21	1.20	1.10	.87	MEAS			

FIGURE 4.12

MCGUIRE-1 CYCLE-2 ASSEMBLY PEAK AXIAL POWERS - CALC (18 LEVEL) VS. MEAS

130 EFPD 100%FP CONTROL BANK D AT 216 STEPS WITHDRAWN

	H	G	F	E	D	C	B	A
	.93	1.05	1.21	.98	.97	.88	1.13	1.25
8	.93	1.04	1.23	.98	1.01	.89	1.14	1.20
	1.07	1.24	1.22	1.03	1.02	.99	1.50	1.27
9	1.07	1.23	1.20	1.01	1.02	.98	1.45	1.20
	1.22	1.22	1.22	1.27	1.03	1.00	1.15	1.17
10	1.23	1.22	1.20	1.26	1.01	1.02	1.14	1.11
	.98	1.03	1.27	1.23	1.24	1.13	1.48	.93
11	.99	1.03	1.26	1.22	1.24	1.11	1.41	.90
	.98	1.03	1.03	1.24	1.34	1.35	1.24	
12	1.03	1.04	1.02	1.24	1.31	1.32	1.19	
	.88	.99	1.00	1.13	1.35	.98	.80	
13	.90	1.00	1.05	1.13	1.34	.95	.76	
	1.13	1.50	1.15	1.48	1.24	.80		
14	1.10	1.43	1.11	1.37	1.19	.77		
	1.25	1.27	1.17	.93	CALC			
15	1.19	1.20	1.10	.87	MEAS			

Q.5 Duke Power Company's contention that no uncertainty in calculated pin powers needs to be accounted for has not been adequately established. One of a set of standard problems, recently developed at Brookhaven National Laboratory for a licensee to assess its ability to calculate typical PWR fuel assemblies, is attached. The licensee's solution using PDQ07 will be an important means of determining the uncertainty in the calculated pin peaking factors.

Re: A.5 Based upon the Duke solution to the BNL benchmark assembly problem, BNL has identified an underprediction of the peak pin power after about 15,000 MWD/MTU which increases to about 1% at 40,000 MWD/MTU. As a result of a conference call held December 11, 1984 between BNL, NRC and Duke, it was determined that a two percent radial local uncertainty was conservative and would be applied in a statistical combination with the reliability factors and engineering hot channel factor.

The three factors to be statistically combined to determine the $F_{\Delta H}^{SCUF}$ factor to multiply the calculated $F_{\Delta H}$ are:

1. F_Q^E, Engineering Heat Flux Hot Channel Factor, is the allowance on heat flux for manufacturing tolerances. This factor allows for local variations in enrichment, pellet density, and diameter. It's numeric value is 1.03.

2. $F_{\Delta H}^R$, the Observed Nuclear Reliability Factor for $F_{\Delta H}$. This factor is developed in Section 11.5 and is 1.03. It represents the ability of EPRI-NODE-P to calculate assembly average power.

3. RL^R, Radial Local Uncertainty or pin power uncertainty. It represents the ability of EPRI-CELL/PDQ07 to calculate the pin power in an assembly. Determined to be 2%.

These factors are statistically combined as follows:

$$F_{\Delta H}^{SCUF} \equiv 1+\sqrt{(.03)^2 + (.03)^2 + (.02)^2} = 1.047.$$

Where SCUF is the statistically combined uncertainty factor.

These factors are statistically independent because they are calculated using different codes and represent different phenomena. The NRC has previously reviewed and approved the statical combination of the radial local uncertainty factor and the $F_{\Delta H}^R$ factor in Northern States Power's report "Qualification of Reactor Physics Methods for Application to Prairie Island Units NSPNAD-8101NP", December 1981. In addition, the NRC has previously reviewed and approved the statistical combination of all three factors in Westinghouse's "Improved Thermal Design Procedure", WCAP-8576, July 1975.

The three factors to be statistically combined to determine the F_Q^{SCUF} factor to multiply the calculated F_Q by are:

1. F_Q^E, Engineering Heat Flux Hot Channel Factor, 1.03.

2. F_Q^R, Assembly Peak Axial Observed Nuclear Reliability Factor. This factor is developed in Section 11.5 and consists of a bias of $(\frac{0.031}{1.375})$ and a $K\sigma$ of 0.048.

3. RL^R, Radial Local Uncertainty or pin power uncertainty, 2%.

The factors are combined to determine the F_Q^{SCUF} factor, where SCUF is the statistically combined uncertainty factor, as follows:

$$F_Q^{SCUF} \equiv 1 + \frac{.031}{1.375} + \sqrt{(.03)^2 + (.048)^2 + (.02)^2} = 1.083$$

$F_{\Delta H}^{SCUF}$ will replace $F_{\Delta H}^R$ in equation 6-2 and F_Q^{SCUF} will replace $F_Q^R \times F_Q^E$ in equation 6-3.

Table 5.1

Benchmark Problem

EPRI-CELL/PDQØ7 Analysis

Maximum Rod Power Summary

Exposure (MWD/MT)	Non-BP Assembly	16-BP Assembly
0	1.060	1.107
500	1.059	1.104
5000	1.054	1.073
10000	1.046	1.041
20000	1.028	1.021
30000	1.014	1.016
40000	1.008	1.010

Table 5.2

Benchmark Problem

Reactivity Defect Calculations

No BP's

		0 MWD/MTU		30000 MWD/MTU	
Case	Description	K-Infinity	% Δρ	K-Infinity	% Δρ
0	Base	1.183699	--	0.896243	-
1	Doppler	1.194852	-0.789	0.907013	-1.325
2	MTC	1.186067	-0.169	0.897301	-0.132
3	68°F	1.211947	-1.969	0.898143	-0.236
4	300°F	1.204695	-1.472	0.904724	-1.046
5	SOLB	1.241994	-3.965	0.937659	-4.928
6	Xe	1.223867	-2.773	0.921068	-3.007
7	Rods	0.789700	42.149	0.605476	53.583

16 BP's

		0 MWD/MTU		30000 MWD/MTU	
Case	Description	K-Infinity	% Δρ	K-Infinity	% Δρ
0	Base	1.020581	--	0.901031	--
1	Doppler	1.030387	-0.932	0.912429	-1.386
2	MTC	1.025619	-0.481	0.903525	-0.306
3	68°F	1.069628	-4.493	0.912266	-1.367
4	300°F	1.053687	-3.079	0.916026	-1.817
5	SOLB	1.060567	-3.694	0.938213	-4.398
6	Xe	1.049333	-2.685	0.926059	-3.000

Table 5.2
(Continued)

Additional Xenon Defect Data

No BP's

	0 MWD/MTU	30,000 MWD/MTU
Xenon Defect (% $\Delta\rho$)	-2.773	-3.007
Xenon Concentration (Atoms/cm^3)[1]	2.1337×10^{15}	1.8623×10^{15}
Xenon Defect (% $\Delta\rho$/atoms/cm^3)[2]	-1.300×10^{-15}	-1.615×10^{-15}

16 BP's

	0 MWD/MTU	30,000 MWD/MTU
Xenon Defect (%	-2.685	-3.000
Xenon Concentration (atoms/cm^3)[1]	2.1334×10^{15}	2.0056×10^{15}
Xenon Defect (% /atoms/cm^3)[2]	-1.259×10^{-15}	1.496×10^{-15}

1. Value averaged over entire assembly volume.
   Fuel to Assembly volume ratio = .90459.

2. Defect per unit volume evaluated over entire assembly.

Table 5.3

1. Name of Codes - PDQØ7; EPRI-CELL[1]
   Code Sources EPRI; EPRI[1]
   Version 2; RAM112[1]

2. Reference for Calculational Method - DPC-NF-2010

3. Assembly Solution Method - Two Group Diffusion Theory

4. Pin-Cell Solution Method - Transport Theory [1]

5. Spatial Mesh Assy/Pin-Cell

   Assembly - One mesh interval per pin

   Pin-Cell[1] - Four Mesh intervals in fuel pin
   One mesh interval in clad
   Five mesh intervals in moderator
   Two mesh intervals in extra region

6. Neutron Cross Section Library - ENDF/B4[1]

7. Number of Fast/Thermal Groups

	No. Fast Groups	No. Thermal Groups
Assembly	1	1
Pin Cell[1]	62	35

8. Depletion Steps -

Assembly (hrs) - 0, 150, 500, 1000, 2000, 3000, 4000, 5000, 6000, 8000, 10000, 12000, 14000, 16000, 18000, 20000, 22000, 24000, 26000, 28000, 30000, 32000, 34000, 36000, 38000, 40000

Pin/Cell(MWD/MTU)[1] 0, 0.001, 100, 500, 1000, 2000, 4000, 6000, 8000, 10000, 12000, 14000, 16000, 18000, 20000, 22000, 24000, 26000, 28000, 30000, 32000, 34000, 36000, 38000, 40000

1 _ All cross-section sets for benchmark problem except CRA and BP were calculated with EPRI-CELL.

Table 5.3 (Continued)

1. Name of Codes - CASMO2E[2]
   Code Sources STUDSVIK
   Version 5

2. Reference for Calculational Method - DPC-NF-2010

3. Assembly Solution Method - Two Group Diffusion Theory

4. Pin-Cell Solution Method - Transport Theory[2]

5. Spatial Mesh Assy/Pin-Cell

   Assembly - One mesh interval per pin[1]

   Pin-Cell - One mesh interval per pin[2]

6. Neutron Cross Section Library - ENDF/B3[2]

7. Number of Fast/Thermal Groups

	No. Fast Groups	No. Thermal Groups
Assembly	4	3
Pin-Cell	9	16

8. Depletion Steps

   Assembly - See Table 5.3 page 1

   Pin-Cell (MWD/MTU)[2] - 0, 150, 500, 1000, 2000, 3000, 5000, 7500, 10000, 12500, 15000, 20000, 25000, 30000, 35000, 40000

[2] - Refers to Burnable Poison and Control Rod Data

Figure A-3. Comparisons of Measured and Predicted Normalized Relative Power Densities for Core 1

INCORE DETECTOR	1.018 1.038 .020	1.011 .997 -.014	.987 .979 -.008	.981 .975 -.006	.997 .978 -.019	.966 .958 -.008	.945 .936 -.009
	1.019 1.035 .016	1.067 1.069 .002	1.012 1.015 .003	1.009 1.012 .003	1.058 1.054 -.004	.999 .988 -.011	.945 .941 -.004
		WATER	1.081 1.087 .006	1.090 1.089 -.001	WATER	1.032 1.045 .013	.953 .947 -.006
			1.054 1.070 .016	1.104* 1.117* .013	1.086 1.100 .014	.989 .994 .005	.945 .939 -.006
				WATER	1.059 1.062 .003	.965 .957 -.008	.934 .928 -.006
					.983 .986 -.002	.938 .937 -.001	.923 .919 -.004
				Measured RPD Calculated RPD ΔRPD		.925 .921 -.004	.914 .911 -.003
							.903 .903 .000

RMS(ΔRPD) = 0.008
Max (ABS(ΔRPD)) = 0.020

*Maximum power fuel rod predicted or measured.

FIGURE 5.1

Figure A-4. Comparisons of Measured and Predicted Normalized Relative Power Densities for Core 5

INCORE DETECTOR	1.005 1.026 .021	.913 .886 -.027	.170 .196 .026	.932 .903 -.029	1.036 1.045 .009	1.063 1.077 .014	1.072 1.090 .018
	.999 1.021 .022	1.017 1.012 -.005	.931 .901 -.030	1.007 .997 -.010	1.125 1.135 .010	1.094 1.112 .018	1.069 1.096 .007
		WATER	.988 .962 -.026	1.087 1.073 -.014	WATER	1.158* 1.174* .016	1.100 1.102 .002
			.181 .203 .022	1.050 1.035 -.015	1.131 1.158 .027	1.038 1.105 .017	1.086 1.090 .004
				WATER	1.048 1.018 -.030	1.035 1.018 -.017	1.070 1.070 .000
					.187 .211 .024	.963 .939 -.024	1.054 1.058 .004
					Measured RPD Calculated RPD ΔRPD	1.018 1.009 -.009	1.060 1.069 .009
							1.070 1.083 .013

UO_2-Gd_2O_3

RMS(ΔRPD) = 0.018

Max (ABS(ΔRPD)) = 0.030

*Maximum power fuel rod predicted or measured.

FIGURE 5.2

PDQØ7 CALCULATED
ROD POWERS

	PDQ-7
PPMB	400
NUMBER BA	0
K-INFINITY	1.18377
BURNUP	0
*MAX. ROD POWER	1.060

0.								
1.035	1.013							
1.038	1.013	1.015						
0.	1.029	1.032	0.					
1.037	1.014	1.018	1.044	1.051				
1.035	1.011	1.015	1.043	1.060 *	0.	PDQØ7		
0.	1.019	1.023	0.	1.042	1.014	0.975		
1.012	0.991	0.993	1.006	0.989	0.961	0.942	0.932	
0.975	0.971	0.970	0.972	0.964	0.951	0.942	0.939	0.949

FIGURE 5.3

PDQØ7 CALCULATED
ROD POWERS

	PDQ-7
PPMB	400
NUMBER BA	0
K-INFINITY	1.17560
BURNUP	500
*MAX. ROD POWER	1.059

0.								
1.035	1.013							
1.037	1.013	1.014						
0.	1.028	1.032	0.					
1.037	1.013	1.017	1.043	1.051				
1.034	1.011	1.015	1.043	1.059 *	0.	PDQØ7		
0.	1.018	1.022	0.	1.041	1.014	.975		
1.012	.992	.993	1.006	.989	.962	.943	.933	
.976	.971	.971	.972	.964	.952	.942	.940	.950

FIGURE 5.4

PDQØ7 CALCULATED
ROD POWERS

	PDQ-7
PPMB	400
NUMBER BA	0
K-INFINITY	1.12604
BURNUP	5000
*MAX. ROD POWER	1.054

0.								
1.032	1.012							
1.034	1.012	1.014						
0.	1.026	1.029	0.					
1.034	1.013	1.016	1.040	1.047				
1.031	1.010	1.014	1.039	1.054 *	0.	PDQØ7		
0.	1.017	1.021	0.	1.038	1.013	.977		
1.011	.992	.993	1.006	.990	.965	.947	.938	
.978	.974	.973	.974	.967	.955	.947	.944	.954

FIGURE 5.5

PDQØ7 CALCULATED
ROD POWERS

	PDQ-7
PPMB	400
NUMBER BA	0
K-INFINITY	1.06962
BURNUP	10.000
*MAX. ROD POWER	1.046

0.								
1.028	1.012							
1.030	1.012	1.013						
0.	1.023	1.026	0.					
1.029	1.012	1.015	1.034	1.040				
1.027	1.010	1.013	1.034	1.046 *	0.	PDQØ7		
0.	1.015	1.018	0.	1.032	1.010	.980		
1.009	.994	.995	1.005	.991	.969	.954	.945	
.981	.978	.977	.978	.971	.961	.953	.950	.958

FIGURE 5.6

PDQØ7 CALCULATED
ROD POWERS

	PDQ-7
PPMB	400
NUMBER BA	0
K-INFINITY	0.97482
BURNUP	20,000
*MAX. ROD POWER	1.028

0.								
1.019	1.010							
1.019	1.010	1.011						
0.	1.016	1.018	0.					
1.019	1.010	1.012	1.023	1.026				
1.017	1.008	1.010	1.022	1.028 *	0.	PDQØ7		
0.	1.010	1.012	0.	1.019	1.006	.986		
1.005	.997	.997	1.002	.994	.980	.969	.962	
.988	.986	.986	.986	.981	.973	.967	.965	.969

FIGURE 5.7

PDQØ7 CALCULATED
ROD POWERS

	PDQ-7
PPMB	400
NUMBER BA	0
K-INFINITY	0.89624
BURNUP	30.000
*MAX. ROD POWER	1.014

0.								
1.010	1.007							
1.011	1.007	1.007						
0.	1.009	1.010	0.					
1.011	1.007	1.008	1.013	1.014				
1.010	1.006	1.007	1.012	1.014 *	0.	PDQØ7		
0.	1.005	1.006	0.	1.009	1.002	.992		
1.002	.999	.999	1.001	.997	.989	.982	.978	
.994	.993	.993	992	.989	.985	.981	.979	.98

FIGURE 5.8

PDQØ7 CALCULATED
ROD POWERS

	PDQ-7
PPMB	400
NUMBER BA	0
K-INFINITY	0.83305
BURNUP	40.000
*MAX. ROD POWER	1.008

0.								
1.005	1.003							
1.005	1.003	1.004						
0.	1.005	1.005	0.					
1.005	1.004	1.004	1.007	1.008 *				
1.005	1.003	1.004	1.006	1.008 *	0.	PDQØ7		
0.	1.003	1.003	0.	1.005	1.001	.996		
1.001	.999	.999	1.000	.998	.994	.991	.988	
.997	.996	.996	.996	.995	.992	.990	.988	.98

FIGURE 5.9

PDQØ7 CALCULATED
ROD POWERS

	PDQ-7
PPMB	400
NUMBER BA	16
K-INFINITY	1.02062
BURNUP	0
*MAX. ROD POWER	1.107

0.								
1.054	1.027							
0.954	0.986	1.029						
0.	0.964	1.046	0.					
0.957	0.986	1.021	1.022	0.959				
1.064	1.030	0.975	0.925	0.883	0.	PDQØ7		
0.	1.060	0.957	0.	0.882	0.906	0.954		
1.107 *	1.062	0.989	0.942	0.950	0.980	1.008	1.038	
1.083	1.066	1.035	1.013	1.013	1.027	1.046	1.067	1.092

FIGURE 5.10

PDQØ7 CALCULATED
ROD POWERS

	PDQ-7
PPMB	400
NUMBER BA	16
K-INFINITY	1.01969
BURNUP	500
*MAX. ROD POWER	1.104

0.								
1.053	1.027							
0.957	0.987	1.028						
0.	0.966	1.046	0.					
0.959	0.987	1.021	1.023	0.962				
1.063	1.030	0.976	0.929	0.888	0.	PDQØ7		
0.	1.059	0.959	0.	0.887	0.910	0.955		
1.104 *	1.059	0.989	0.944	0.952	0.980	1.006	1.034	
1.080	1.063	1.033	1.012	1.012	1.025	1.042	1.062	1.08

FIGURE 5.11

PDQØ7 CALCULATED
ROD POWERS

	PDQ-7
PPMB	400
NUMBER BA	16
K-INFINITY	1.02749
BURNUP	5.000
*MAX. ROD POWER	1.073

0.								
1.046	1.023							
0.983	0.997	1.024						
0.	0.987	1.040	0.					
0.984	0.996	1.019	1.027	0.987				
1.052	1.024	0.989	0.963	0.938	0.	PDQØ7		
0.	1.045	0.979	0.	0.933	0.941	0.961		
1.073 *	1.038	0.991	0.964	0.964	0.975	0.988	1.005	
1.047	1.035	1.014	1.000	0.998	1.003	1.012	1.025	1.044

FIGURE 5.12

PDQØ7 CALCULATED
ROD POWERS

	PDQ-7
PPMB	400
NUMBER BA	16
K-INFINITY	1.02238
BURNUP	10,000
*MAX. ROD POWER	1.041

0.								
1.036	1.018							
1.007	1.005	1.018						
0.	1.006	1.031	0.					
1.007	1.004	1.016	1.027	1.008				
1.038	1.016	1.000	0.995	0.986	0.			
0.	1.029	0.998	0.	0.978	0.972	0.970		
1.041 *	1.017	0.994	0.984	0.977	0.973	0.974	0.979	
1.017	1.009	0.998	0.990	0.986	0.984	0.986	0.992	1.000

PDQØ7

FIGURE 5.13

PDQØ7 CALCULATED
ROD POWERS

	PDQ-7
PPMB	400
NUMBER BA	16
K-INFINITY	0.97150
BURNUP	20.000
*MAX. ROD POWER	1.021

0.								
1.020	1.010							
1.018	1.009	1.010						
0.	1.015	0.017	0.					
1.018	1.008	1.010	1.020	1.020				
1.019	1.008	1.007	1.017	1.021 *	0.	PDQØ7		
0.	1.012	1.010	0.	1.012	1.000	0.984		
1.010	1.000	0.997	1.000	0.991	0.978	0.971	0.968	
0.994	0.991	0.988	0.987	0.982	0.976	0.973	0.973	0.97[illegible]

FIGURE 5.14

PDQØ7 CALCULATED
ROD POWERS

	PDQ-7
PPMB	400
NUMBER BA	16
K-INFINITY	0.90103
BURNUP	30,000
*MAX. ROD POWER	1.016

0.								
1.011	1.006							
1.013	1.007	1.006						
0.	1.011	1.010	0.					
1.012	1.007	1.007	1.012	1.014				
1.010	1.005	1.006	1.013	1.016 *	0.	PDQØ7		
0.	1.005	1.007	0.	1.010	1.002	0.991		
1.003	1.000	1.000	1.001	1.000	0.987	0.981	0.978	
0.995	0.993	0.992	0.991	0.988	0.984	0.981	0.980	0.98

FIGURE 5.15

PDQØ7 CALCULATED
ROD POWERS

	PDQ-7
PPMB	400
NUMBER BA	16
K-INFINITY	0.84163
BURNUP	40.000
*MAX. ROD POWER	1.010

0.								
1.005	1.003							
1.007	1.004	1.004						
0.	1.006	1.005	0.					
1.007	1.004	1.004	1.007	1.008				
1.005	1.003	1.004	1.008	1.010 *	0.	PDQØ7		
0.	1.003	1.004	0.	1.007	1.002	0.995		
1.001	0.999	0.999	1.001	0.998	0.993	0.989	0.987	
0.996	0.996	0.995	0.995	0.993	0.990	0.988	0.988	0.989

FIGURE 5.16

Q.6 Please provide the updates to DPC-NF-2010, if any, that will make it consistent with the methodologies being used by Duke Power.

A.6 The following sections address updates to the methods described in DPC-NF-2010.

1. EPRI-NODE-P Normalization:

In addition to adjusting radial albedoes, small M^2 adjustments are made for various fuel types (usually only fresh fuel) to attain better agreement with PDQØ7 radial power calculations. Fugures 6.1 and 6.2 show the improvement for assembly radial powers with respect to measurement. Figures 6.3 and 6.4 address assembly peak power improvements. The data in figures 6.1 through 6.4 represent McGuire Unit 1 Cycle 2.

2. Axial Nodal Modeling:

Section 11 of DPC-NF-2010 presents a benchmark analysis which employed twelve axial nodes per assembly. Core-specific axial modeling would conform to the physics requirements of the core. Answer 4 addressed the calculated-to-measured improvement shown by employing eighteen axial nodes per assembly. Should future fuel assemblies become non-uniform, i.e., axial blankets or part length burnable absorbers, the Duke Power version of EPRI-NODE-P can adequately model the core.

Since the upgrades described in parts 1 and 2 have significantly improved calculated-to-measured agreement, the ONRF values for F_Q and $F_{\Delta H}$ in DPC-NF-2010 are considered conservative. Therefore, even though the upgraded methods have demonstrated improved agreement, Duke Power will still employ previously derived ONRFs.

3. EPRI-NODE-P Enhancements:

EPRI-NODE-P has received several major enhancements which are discussed below. This enhanced version was used throughout the analyses shown in DPC-NF-2010. These enhancements are:

a. Partial reactivity formulations due to xenon, moderator temperature, and doppler temperature have been revised to include third order burnup dependent multipliers.

b. Fuel assemblies can be axially modeled as containing up to three different fuel types.

c. Nodded M^2 is linearly adjusted according to the fraction of node length occupied by a control rod.

d. The full power volumetric average fuel temperature has been revised to a burnup dependent fourth order polynomial.

f. The nodal source convergence routine has been modified to use the Gauss-Seidel iterative method with the inclusion of an optional acceleration parameter.

g. Minor enhancements have also been made which allow more user-friendly input and output features.

Likewise, Duke Power's fitting code EPRI-SUPERLINK has been modified to provide compatibility with EPRI-NODE-P. All codes are rigorously tested and certified before production usage in conformance with Duke Power's Q/A procedures.

MCGUIRE-1 CYCLE-2 ASSEMBLY RADIAL POWERS - CALC (NO MSQUARE ADJ) VS. MEAS

48 EFPD 100%FP CONTROL BANK D AT 228 STEPS WITHDRAWN

	H	G	F	E	D	C	B	A
	.84	.97	1.15	.88	.88	.73	.91	1.03
8	.83	.94	1.11	.87	.90	.77	.98	1.06
	.99	1.18	1.16	.93	.90	.83	1.25	1.04
9	.97	1.12	1.10	.91	.91	.85	1.26	1.05
	1.16	1.16	1.16	1.20	.92	.86	.93	.96
10	1.13	1.11	1.11	1.16	.91	.89	.99	.97
	.88	.94	1.20	1.16	1.15	.97	1.23	.76
11	.89	.93	1.16	1.11	1.12	.98	1.23	.77
	.89	.91	.92	1.15	1.23	1.16	1.05	
12	.91	.92	.91	1.13	1.19	1.19	1.06	
	.74	.84	.86	.97	1.16	.79	.65	
13	.79	.87	.91	.99	1.20	.83	.66	
	.91	1.25	.93	1.24	1.05	.65		
14	.98	1.26	.97	1.20	1.05	.67		
	1.03	1.05	.96	.76	CALC			
15	1.06	1.05	.95	.75	MEAS			

FIGURE 6.1

MCGUIRE-1 CYCLE-2 ASSEMBLY RADIAL POWERS - CALCULATED VS. MEASURED

48 EFPD 100%FP CONTROL BANK D AT 228 STEPS WITHDRAWN

	H	G	F	E	D	C	B	A
8	.83 .83	.96 .94	1.10 1.11	.88 .87	.86 .90	.75 .77	.95 .98	1.03 1.06
9	.98 .97	1.13 1.12	1.12 1.10	.93 .91	.91 .91	.85 .85	1.24 1.26	1.04 1.05
10	1.11 1.13	1.12 1.11	1.13 1.11	1.16 1.16	.93 .91	.88 .89	.97 .99	.96 .97
11	.88 .89	.94 .93	1.16 1.16	1.13 1.11	1.13 1.12	1.00 .98	1.23 1.23	.75 .77
12	.86 .91	.91 .92	.93 .91	1.13 1.13	1.22 1.19	1.19 1.19	1.04 1.06	
13	.76 .79	.86 .87	.88 .91	1.00 .99	1.19 1.20	.85 .83	.66 .66	
14	.96 .98	1.24 1.26	.98 .97	1.23 1.20	1.05 1.05	.66 .67		
15	1.03 1.06	1.05 1.05	.96 .95	.75 .75	CALCULATED MEASURED			

FIGURE 6.2

MCGUIRE-1 CYCLE-2 ASSEMBLY RADIAL POWERS - CALCULATED VS. MEASURED

48 EFPD 100%FP CONTROL BANK D AT 228 STEPS WITHDRAWN

	H	G	F	E	D	C	B	A
8	.83 .83	.96 .94	1.10 1.11	.88 .87	.86 .90	.75 .77	.95 .98	1.03 1.06
9	.98 .97	1.13 1.12	1.12 1.10	.93 .91	.91 .91	.85 .85	1.24 1.26	1.04 1.05
10	1.11 1.13	1.12 1.11	1.13 1.11	1.16 1.16	.93 .91	.88 .89	.97 .99	.96 .97
11	.88 .89	.94 .93	1.16 1.16	1.13 1.11	1.13 1.12	1.00 .98	1.23 1.23	.75 .77
12	.86 .91	.91 .92	.93 .91	1.13 1.13	1.22 1.19	1.19 1.19	1.04 1.06	
13	.76 .79	.86 .87	.88 .91	1.00 .99	1.19 1.20	.85 .83	.66 .66	
14	.96 .98	1.24 1.26	.98 .97	1.23 1.20	1.05 1.05	.66 .67		
15	1.03 1.06	1.05 1.05	.96 .95	.75 .75	CALCULATED MEASURED			

FIGURE 6.2

MCGUIRE-1 CYCLE-2 ASSEMBLY PEAK AXIAL POWERS - CALC (NO MSQUARE ADJ) VS. MEAS

48 EFPD 100%FP CONTROL BANK D AT 228 STEPS WITHDRAWN

	H	G	F	E	D	C	B	A
8	.97 .92	1.11 1.04	1.29 1.23	1.00 .97	1.00 1.00	.81 .86	1.06 1.16	1.30 1.25
9	1.13 1.07	1.32 1.24	1.30 1.22	1.06 1.01	1.02 1.01	.93 .97	1.54 1.48	1.32 1.24
10	1.30 1.25	1.31 1.23	1.31 1.22	1.33 1.27	1.03 1.00	.95 1.01	1.09 1.17	1.20 1.14
11	1.01 .99	1.07 1.03	1.33 1.26	1.28 1.22	1.28 1.25	1.10 1.12	1.53 1.45	.94 .90
12	1.01 1.00	1.02 1.02	1.03 1.00	1.28 1.25	1.41 1.34	1.37 1.37	1.30 1.23	
13	.82 .87	.93 .97	.95 1.04	1.10 1.14	1.37 1.39	.92 .96	.78 .76	
14	1.06 1.11	1.55 1.45	1.10 1.12	1.54 1.39	1.30 1.23	.79 .78		
15	1.30 1.26	1.32 1.25	1.21 1.12	.95 .87	CALCULATED MEASURED			

FIGURE 6.3

MCGUIRE-1 CYCLE-2 ASSEMBLY PEAK AXIAL POWERS - CALC (1B LEVEL) VS. MEAS

48 EFPD 100ZFP CONTROL BANK D AT 228 STEPS WITHDRAWN

	H	G	F	E	D	C	B	A
8	.96 .92	1.10 1.04	1.25 1.23	1.00 .97	.98 1.00	.84 .86	1.11 1.16	1.28 1.25
9	1.12 1.07	1.28 1.24	1.26 1.22	1.06 1.01	1.03 1.01	.95 .97	1.51 1.48	1.31 1.24
10	1.26 1.25	1.27 1.23	1.27 1.22	1.30 1.27	1.04 1.00	.97 1.01	1.14 1.17	1.19 1.14
11	1.01 .99	1.07 1.03	1.30 1.26	1.26 1.22	1.26 1.25	1.14 1.12	1.52 1.45	.93 .90
12	.98 1.00	1.03 1.02	1.04 1.00	1.26 1.25	1.39 1.34	1.40 1.37	1.28 1.23	
13	.84 .87	.95 .97	.97 1.04	1.14 1.14	1.40 1.39	.98 .96	.79 .76	
14	1.11 1.11	1.52 1.45	1.15 1.12	1.52 1.39	1.28 1.23	.80 .78		
15	1.29 1.26	1.31 1.25	1.20 1.12	.93 .87	CALC MEAS			

FIGURE 6.4

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

June 26, 2002

Mr. M. S. Tuckman
Executive Vice President
Nuclear Generation
Duke Energy Corporation
526 South Church St
Charlottte, NC 28202

SUBJECT: CATAWBA NUCLEAR STATION, UNITS 1 AND 2 AND MCGUIRE NUCLEAR STATION, UNITS 1 AND 2 RE: REQUEST FOR ADDITIONAL INFORMATION - APPLICATION FOR CHANGES TO TECHNICAL SPECIFICATIONS (TAC NOS. MB3343, MB3344, MB3222 AND MB3223)

Dear Mr. Tuckman:

The Nuclear Regulatory Commission is reviewing your application dated October 7, 2001, entitled "License Amendment Request applicable to Technical Specifications 5.6.5, Core Operating Limits Report; Revisions to Bases 3.2.1 and 3.2.3; and Revisions to Topical Reports DPC-NE-2009-P, DPC-NF-2010, DPC-NE-2011-P, and DPC-NE-1003" and has identified a need for additional information as identified in the Enclosure. These issues were discussed with your staff on June 6, 2002. Please provide a response to this request within forty-five (45) days of receipt of this letter so that we may complete our review.

Sincerely,

Robert Martin

Robert E. Martin, Senior Project Manager, Section 1
Project Directorate
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket Nos. 50-413, 50-414, 50-369 and 50-370

Enclosure: Request for Additional Information

cc w/encl: See next page

cc:

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REQUEST FOR ADDITIONAL INFORMATION

LICENSE AMENDMENT REQUEST APPLICABLE TO

TECHNICAL SPECIFICATION 5.6.5, CORE OPERATING LIMITS REPORT,

REVISIONS TO BASES 3.2.1 and 3.2.3

REVISIONS TO TOPICAL REPORTS DPC-NE-2009-P,

DPC-NF-2010, DPC-NE-2011-P, AND DPC-NE-1003

CATAWBA NUCLEAR STATION, UNITS 1 AND 2

MCGUIRE NUCLEAR STATION, UNITS 1 and 2

DUKE ENERGY CORPORATION

Topical Reports Numbered DPC-NE-2009-P Duke Power Company Westinghouse Fuel Transition Report and DPC-NF-2010-A, Duke Power Company McGuire Nuclear Station and Catawba Nuclear Station Nuclear Physics Methodology for Reload Design

1. Please provide a detailed qualitative technical justification for the requested changes to the topical reports (methodologies), DPC-NE-2011 and DPC-NF-2010. (i.e., why are these changes being made?).

2. To expedite the review process, please pro[illegible]de a qualitative and quantitative technical basis for each of the changes in these top[illegible]rts.

3. Please provide validation data that bench-marks the results of comparisons between the old and the new models (changes).

4. If the changes to these topical reports and methodologies impact the safe operation of the reactor core, please provide the safety significance (impact) of each of these changes.

5. Please provide the basis for why the proposed changes to the above stated topical reports should be found acceptable.

Topical Report Numbered DPC-NF-2010-A, Duke Power Company McGuire Nuclear Station and Catawba Nuclear Station Nuclear Physics Methodology for Reload Design

1. In the revision history section on page ii, the licensee provides the staff with the reason for the submittal. Since this is a licensing action, please list those Technical Specification(s), Bases, FSAR sections, conformance to regulatory documents, criteria, generic letters, etc. that are impacted by the request for these changes within the licensing framework.

2. Section 4.2.4.2, second paragraph. Please provide clarification of this change and the technical justification for it. Please provide a comparison between the old sentence and the new sentence.

3. In Attachment 7a, "Detailed Listing of the Changes to DPC-NF2010A," it is stated in many places, that "this change is made to avoid difficulties with the literal interpretation of the original description." Please provide clarification of this statement with a supporting example.

4. Section 4.2.4.4, fifth paragraph. Please provide clarification of this change and the technical justification for it. Please provide comparison between the old sentence and the new sentence.

5. Section 8.1, first paragraph. Is the added equation the same as that in the current version of the DPC-NF-2010A topical? If not, please provide technical justification for its use.

6. Section 9.1.5, first paragraph. Please provide clarification of this change and the technical justification for it. Please provide a comparison between the old sentence and the new sentence.

Topical Report Numbered DPC-NE-2011-P-A, Duke Power Company Nuclear Design Methodology Report for Core Operating Limits of Westinghouse Reactors

1. The description of the transient conditions was changed in Tables 1 and 2, of Section 2.5. It is not clear to the staff exactly what was changed. Please clarify.

2. From section 6.1, please explain what is meant by "updated the equation."

3. From section 6.1, please provide further clarification of this statement.

4. Section 6.2, were is UMR listed in section 6.2? Please provide original definition and new definition for comparison.

Topical Report Numbered DPC-NE-1003, Revision 1 McGuire Nuclear Station and Catawba Nuclear Station Rod Swop Methodology Report for Startup Physics Testings, Revision 1

1. Appendix A of topical report DPC-NE-1003, Revision 1, contains two versions of Duke Power Company's rod swap measurement procedure PT/O/A/4150/11A: Attachment 3 (dated June 1986) and Attachment 4 (dated April 1984). There are differences in these two versions of the procedure. For example, in the Attachment 3 version, Steps 12.2.2 and 12.2.3, respectively, specify the insertion of bank 1 until the indicated reactivity is approximately -20 pcm, and the withdrawal of reference bank until the indicated reactivity is approximately +20 pcm; whereas in the Attachment 4 version, the insertion and withdrawal of bank 1 and reference bank, respectively, of steps 12.2.1 and 12.2.2 specify reactivity change of -/+ 10 pcm.

a. Since the Attachment 3 version of procedures is more recent, why is the Attachment 4 version referenced in Revision 1 of the topical report (Reference 2)?

b. Which of these two versions of rod swap measurement procedures will be used for McGuire and Catawba Units?

2. In the Attachment 3 version of rod swap measurement procedures PT/O/A/4150/11A, Step 12.1.3 states that: "Repeat steps 12.2.1 and 12.2.2 until the previously inserted bank is fully withdrawn."

   Is there a typographic error in the words "steps 12.2.1 and 12.2.2"? Should correct words be "steps 12.1.1 and 12.1.2"?

3. The equation in Section 3, Measurement Procedure, of the topical report for calculating the inferred rod worth of bank x is different from the equation in Step 12.5.3 of the Attachment 3 procedures. The difference appears to be due to the initial height of the reference bank for performing the rod swap measurement of the measured bank.

   Clarify the exact procedure to be used in the rod swap test, and make all necessary corrections in the topical report and the procedures to be consistent.

4. The third sentence in Section 3 of the topical report is revised to read: "All other banks are then exchanged with the reference bank or other test banks at constant boron conditions until the measured bank is fully inserted." It is stated, in Attachment 9a, "Detailed Listing of Changes to DPC-NE-1003A," that the third sentence in Section 3 is revised to make the report consistent with current procedures. The "Revision History" in the topical report states that this revision [Revision 1] also reflects a refinement in the rod swap to make use of two test banks.

   a. What are the current procedures? What is the date of the current procedures?

   b. Are the current procedures the same or different from the ones in Attachment 3? The Attachment 3 procedures do not include the exchange of a test bank with the other test bank.

   c. If the current procedures are different from those of Attachment 3 or 4, provide a copy of the procedures, and appropriately reference them in the report.

   d. Is the statement in "Revision History" referring to this revision? Please explain what the statement means.

Duke Power
A Duke Energy Company

Duke Energy Corporation
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Michael S. Tuckman
Executive Vice President
Nuclear Generation

August 7, 2002

U. S. Nuclear Regulatory Commission
Washington D.C. 20555-0001
ATTENTION: Document Control Desk

Subject: Duke Energy Corporation

McGuire Nuclear Station, Units 1 and 2
Docket Nos. 50-369 and 370

Catawba Nuclear Station, Units 1 and 2
Docket Nos. 50-413 and 414

Response to NRC Request for Additional Information - TAC nos. MB3222, MB3223, MB3343, and MB3344) and License Amendment Request Supplement

This purpose of this letter is to provide Duke Energy Corporation's (Duke) response to an NRC request for additional information (RAI) and to supplement a Duke license amendment request (LAR) previously submitted pursuant to 10CFR50.90. Please note that some of the information contained in this submittal package has been determined to be proprietary and is being submitted pursuant to 10CFR2.790. This proprietary information is discussed below.

Duke submitted[1] a LAR applicable to McGuire and Catawba Technical Specifications (TS) 5.6.5.a and 5.6.5.b. Also included in this submittal were proposed revisions to the four Duke Topical Reports listed below.

[1] Reference 1: Letter, Duke Energy Corporation to U.S. Nuclear Regulatory Commission, ATTENTION: Document Control Desk, Dated October 7, 2001, SUBJECT: License Amendment Request Applicable to Technical Specification 5.6.5, Core Operating Limits Report; Revisions to Bases 3.2.1 and 3.2.3; and Revisions to Topical Reports DPC-NE-2009-P, DPC-NF-2010, DPC-NE-2011-P, and DPC-NE-1003

- DPC-NE-2009-P, *Duke Power Company Westinghouse Fuel Transition Report*, Revision 1;

- DPC-NF-2010, *Duke Power Company McGuire Nuclear Station and Catawba Nuclear Station Nuclear Physics Methodology for Reload Design*, Revision 1;

- DPC-NE-2011-P, *Duke Power Company Nuclear Design Methodology Report for Core Operating Limits of Westinghouse Reactors*, Revision 1;

- DPC-NE-1003, McGuire Nuclear Station and Catawba Nuclear Station *Rod Swap Methodology Report for Startup Physics Testing*, Revision 1.

The NRC RAI[2] asked questions on these topical reports. As described below, the Duke responses to these questions are included in the attachments to this letter.

In a subsequent submittal,[3] Duke proposed another LAR for McGuire and Catawba TS 5.6.5, but this LAR was only applicable to TS 5.6.5.b. The information contained herein explains the necessary coordination for changing TS 5.6.5.b for McGuire and Catawba. This LAR implements the provisions of an NRC approved Technical Specifications Task Force (TSTF) Standard Technical Specifications Traveler.[4] The NRC has approved and issued this LAR for both McGuire[5] and Catawba.[6] Implementation of the

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[2] Reference 2: Letter, U. S. Nuclear Regulatory Commission to Duke Energy Corporation, Dated June 26, 2002, SUBJECT: Request for Additional Information, Application for Changes to Technical Specifications (TAC Nos. MB3222, MB3223, MB3343, and MB3344

[3] Reference 3, Letter, Duke Energy Corporation to U.S. Nuclear Regulatory Commission, ATTENTION: Document Control Desk, Dated December 20, 2001, SUBJECT: License Amendment Request Applicable to the Technical Specifications Requirements for the Core Operating Limits Report – Oconee, McGuire, and Catawba Technical Specification 5.6.5

[4] TSTF-363, "Revise Topical Report References in ITS 5.6.5 COLR"

[5] Letter, U. S. Nuclear Regulatory Commission to Duke Energy Corporation Dated July 10, 2002, SUBJECT: McGuire Nuclear Station, Units 1 and 2 RE: Issuance of Amendments (TAC Nos. MB3702 and MB3703)

[6] Letter, U. S. Nuclear Regulatory Commission to Duke Energy Corporation Dated July 2, 2002, SUBJECT: Catawba Nuclear Station, Units 1 and 2 RE: Issuance of Amendments (TAC Nos. MB3728 and MB3729)

referenced industry traveler eliminates the need for the changes Duke proposed to McGuire and Catawba TS 5.6.5.b in Reference 1. The LAR supplement transmitted herein deletes the proposed changes to McGuire and Catawba TS 5.6.5.b contained in Reference 1. The attached McGuire and Catawba TS pages (both marked and reprinted versions) update Reference 1 such that it contains the latest approved version of the affected TS pages and only applies to McGuire and Catawba TS 5.6.5.a. The affected TS pages are:

McGuire Units 1 and 2 Pages: 5.6-2, 5.6-3, B3.2.1-11, and B3.2.3-4; and

Catawba Units 1 and 2 Pages: 5.6-3, B3.2.1-11, and B3.2.3-4.

As shown, conforming Bases changes have been made and the necessary Bases pages are also included.

The attachments to this letter are listed and described below.

- Attachment 1 provides the Duke response to the NRC's general questions on Topical Reports DPC-NF-2010 and DPC-NE-2011-P.

- Attachment 2 provides the Duke response to the NRC's specific questions on Topical Report DPC-NF-2010.

- Attachments 3a and 3b provide the Duke responses to the NRC's specific questions on Topical Report DPC-NE-2011-P. Attachment 3a is the proprietary version and Attachment 3b is the non-proprietary version.

- Attachment 4 provides the Duke response to the NRC's specific questions on Topical Report DPC-NE-1003.

- Attachment 5 provides the Duke response to an NRC concern on Topical Report DPC-NE-2009-P. This concern was not included in the NRC's RAI,[2] however it was discussed during an NRC/Duke telephone conference held on July 24, 2002.

- Attachments 6a and 6b provide a marked copy of the existing approved Technical Specifications pages for McGuire Units 1 and 2 and Catawba Units 1 and 2, respectively. These marked copies show the proposed changes.

- Attachments 7a and 7b provide the reprinted Technical Specifications and Bases pages for McGuire Units 1 and 2 and Catawba Units 1 and 2, respectively.

Duke has determined that the revisions contained in this LAR supplement, as shown in Attachments 6a, 6b, 7a, and 7b have no impact on the determination of no significant hazards consideration that was included in Reference 1.

This submittal package contains information that Duke considers proprietary. This information is contained within the proprietary version of the response to the NRC questions on Topical Report DPC-NE-2011-P that is provided as Attachment 3a to this letter. In accordance with 10CFR2.790, Duke requests that this information be withheld from public disclosure. An affidavit that attests to the proprietary nature of this information is included with this letter. A non-proprietary version of this response is also provided as Attachment 3b to this letter.

Inquiries on this matter should be directed to J. S. Warren at (704) 382-4986.

Very truly yours,

M. S. Tuckman

M. S. Tuckman

xc w/Attachments:

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South Carolina Bureau of Land and Waste Management
2600 Bull Street
Columbia, SC 29201

M. S. Tuckman, affirms that he is the person who subscribed his name to the foregoing statement, and that all the matters and facts set forth herein are true and correct to the best of his knowledge.

M. S. Tuckman

M. S. Tuckman, Executive Vice President

Subscribed and sworn to me: August 7, 2002
Date

Mary P. Dehus, Notary Public

My commission expires: JAN 22, 2006

SEAL

bxc w/Attachments:

M. T. Cash
C. J. Thomas
G. D. Gilbert
L. E. Nicholson
K. L. Crane
K. E. Nicholson
J. M. Ferguson (2) - CN01SA
L. J. Rudy
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R. L. Gill
P. M. Abraham
G. G. Pihl
D. R. Koontz
R. C. Harvey
MNS Master File - MG01DM
Catawba Master File - CN04DM
NRIA/ELL

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Laurens, SC 29360-0929

NC Municipal Power Agency No. 1
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Raleigh, NC 27626-0513

T. R. Puryear
NC Electric Membership Corporation
CN03G

Piedmont Municipal Power Agency
121 Village Drive
Greer, SC 29651

Attachment 1

Responses to Request for Additional Information

Topical Reports Numbered DPC-NE-2011-P, Revision 1, Duke Power Company Nuclear Design Methodology Report for Core Operating Limits of Westinghouse Reactors and DPC-NF-2010, Revision 1, Duke Power Company McGuire Nuclear Station and Catawba Nuclear Station Nuclear Physics Methodology for Reload Design (TAC NOS. MB3343, MB3344, MB3222, MB3223)

General

Subsequent to receiving the NRC RAI package, a clarification of Questions 1, 2, and 3 was obtained from the NRC during a conference call on Thursday July 18, 2002. Responses to all questions in the NRC RAI are given below, and responses to Questions 1, 2, and 3 take into account the clarification received from the NRC.

Question 1. Please provide a detailed qualitative technical justification for the requested changes to the topical reports (methodologies), DPC-NE-2011 and DPC-NF-2010. (i.e., why are these changes being made?).

Response

Subsequent to the approval of the current version of these reports, there have been various changes in calculation methods and plant operating philosophy. Therefore, sections of these topical reports affected by these changes have been reviewed and updated to improve clarity and continuity in order to avoid ambiguities and inconsistencies that could be misconstrued. These revisions do not change approved methods nor introduce new methods. These changes and justifications were identified and described in the October 7, 2001 DEC submittal.

Question 2. To expedite the review process, please provide a qualitative and quantitative technical basis for each of the changes in the above stated topical reports.

Response

Qualitative and quantitative bases for each change to DPC-NF-2010 and DPC-NE-2011-P are provided in Attachments 7a and 8a, respectively in the License Amendment Request package submitted by Duke with a cover letter date of October 7, 2001.

Question 3. Please provide validation data, bench-marking the results of comparisons between the old and the new models (changes).

Response

These revisions do not change approved methods nor introduce new methods; therefore, additional benchmarking is not necessary.

Question 4. If the changes to these topical reports/methodologies impact the safe operation of the reactor core, please provide the safety significance (impact) of each of these changes?

Response

The methodology changes correspond to previously approved methodologies or licensing basis documents, or to administrative non-technical changes. Therefore, these changes do not impact the safe operation of the reactor core.

Question 5. Please provide the basis as to why the proposed changes to the above stated topical reports should be found acceptable.

Response
The purpose for these changes is to maintain the topical reports in a condition that is consistent with other current, NRC approved licensing related documents and to improve clarity and continuity in order to avoid ambiguities and inconsistencies that could be misconstrued. The changes do not change previously approved methodologies.

Attachment 2
Responses to Request for Additional Information
Topical Report Numbered DPC-NF-2010, Revision 1, Duke Power Company McGuire Nuclear Station and Catawba Nuclear Station Nuclear Physics Methodology for Reload Design (TAC NOS. MB3343, MB3344, MB3222, MB3223)

Question 1. In the revision history section on page ii, the licensee provides the staff with the reason for the submittal. Since this is a licensing action, please list/Tabulate what Technical Specification(s), Basis, FSAR, conformance to regulatory documents, criteria, generic letters, etc., etc. are impacted by the request for these changes within the licensing framework?

Response
The impact to licensing basis documents by changes made to DPC-NF-2010 is described below.

- Technical Specifications and Bases: TS 5.6.5.b

No Technical Specification or Bases requires a change as a result of these revisions. Even the Licensing Amendment Request to change Technical Specification 5.6.5b for this proposed topical report revision is no longer required (see the License Amendment Request to implement the provisions of an NRC approved Technical Specifications Task Force (TSTF) Standard Technical Specifications Traveler (TSTF 363, "Revise Topical Report References in ITS 5.6.5 COLR")).

- UFSAR Sections: 1.6.3, 4.3, and 15.0
- Topical Reports: DPC-NE-1004, DPC-NE-1003, DPC-NE-2004P, DPC-NE-2007P DPC-NE-2009P, DPC-NE-3001P

These documents contain general references to the methods contained in the proposed topical report. Changes to these documents are expected to be made as part of the normal UFSAR and Topical Report update processes.

Question 2. Section 4.2.4.2, second paragraph. Please provide clarification of this change and the technical justification for it. Please provide comparison between the old sentence and the new sentence.

Response
Original Sentence: "Cases are run with the moderator temperature at 5 °F above and at the reference temperatures."

Proposed Sentence: "Cases are run changing the moderator temperature from the reference temperature."

The original sentence may imply that the calculation of the moderator temperature coefficient will be performed by only changing the moderator temperature +5 °F. Whereas, these calculations may be more appropriately performed using a -5 °F change, using an average of the +5 and -5 °F results, or using a different temperature change depending on actual plant conditions. Therefore, specificity is removed to reflect that calculations are performed to match plant conditions or intended use of the data.

Question 3. In Attachment 7a–Detailed Listing of the Changes to DPC-NF2010A, it is stated in many places, that "this change is made to avoid difficulties with the literal interpretation of the original description". Please provide clarification of this statement with a supporting example.

Response
Changes documented in Attachment 7a which state "this change is made to avoid difficulties with the literal interpretation of the original description" also provide additional information about the reason why the literal interpretation could potentially be misconstrued. Changes with this statement can be categorized into 3 types: (1) descriptions of plant operations, (2) descriptions of calculations, and (3) administrative. An example within each category is provided below.

Descriptions of Plant Operations
Example: Change #3

Section 1.1, First Paragraph
Description: Changed the third sentence to give examples of intervals between refueling outages.
Justification: The original sentence implies a maximum fuel cycle length of 18 months, and possible fuel cycle lengths are not limited to 18 months. This change is made to avoid difficulties with the literal interpretation of the original description.

The current version states: "Refueling occurs at intervals of 6 to 18 months, depending on the utility's operational requirements."

The proposed version states: "Refueling occurs at intervals appropriate for the power production needed, for example 12, 18, or 24 months."

A literal interpretation of the current version may imply that development of a core design is limited to a 6 to 18 month fuel cycle, whereas current core designs may be different from the exact range of 6 to 18 months.

Descriptions of Calculations
Example: Change #32

Section 4.2.1, Third Paragraph
Description: Clarified the first sentence.
Justification: Depletion model statepoints may be specified in MWD/MTU or EFPD and may be different than those listed. This change is made to avoid difficulties with the literal interpretation of the original description.

The current version states: "The cycle is then depleted in steps corresponding to 0, 150, 500, 1000, 2000, 4000 ... MWD/MTU to verify that power peaking versus burnup remains acceptable."

The proposed version states: "The cycle is then depleted to various times in the cycle to verify that power peaking versus burnup remains acceptable."

A literal interpretation of the current version may imply that core depletions would have to be performed at the burnup statepoints listed, using MWD/MTU units, and at specific burnup intervals. Current core depletions may use a different set of burnup statepoints and intervals

depending on fuel and burnable poison depletion effects. Also, burnup statepoints may be specified in units other than MWD/MTU (for example EFPD).

Administrative
Example: Change #104

Section 9.1.2, First Paragraph
Description: Changed the last sentence for clarity.
Justification: This change is made to avoid difficulties with the literal interpretation of the original description. Equilibrium xenon worth data may be shown in plot or table format.

The current version states: "The results are displayed in a format similar to Figure 9-4."

The proposed version states: "Figure 9-4 shows the results of a typical equilibrium xenon worth calculation."

A literal interpretation of the current version may imply that equilibrium xenon worth calculation results would be displayed in a plot format to be used in startup test predictions and core physics parameters. However, it is also acceptable to provide this information in a table or electronic database.

Question 4. Section 4.2.4.4, fifth paragraph. Please provide clarification of this change and the technical justification for it. Please provide comparison between the old sentence and the new sentence.

Response
Original Sentence: "Then a second EPRI-NODE case is run with the core power level reduced 5% while holding everything else constant."

Proposed Sentence: "Then a second case is run with the core power reduced while holding control rods, boron, and xenon constant."

The original sentence may imply that the calculation of the power coefficient will be performed by changing the core power -5%. Whereas, these calculations may be more appropriately performed using a different power reduction or increase depending on actual plant conditions. Therefore, specificity is removed to reflect that calculations are performed to match plant conditions or intended use of the data. By removing the reference to the core simulator, the implication is made that any NRC approved model may be used. Finally, the revised sentence removes the ambiguity of the statement "everything else".

Question 5. Section 8.1, first paragraph. Is the added equation the same as that in the current version of the DPC-NF-2010A topical? If not, please provide technical justification for its use.

Response
The equation is in the current approved version of DPC-NF-2010. This equation is located in Section 6.2.1.2 (Page 6-2) of the current version and is labeled Equation "6-1". Section 6 of the proposed version was rewritten for reasons explained in Attachment 7a of the Licensing Amendment Request Package dated October 7, 2001.

Section 6 was rewritten, because subsequent to the initial NRC approval of this topical report, methods for performing safety related calculations were approved by the NRC in References 1, 2, and 3 (below). The NRC excluded Section 6.3 when the NRC SER of the original version of this report was issued. The rewrite of this section references safety analysis methods approved by the NRC (References 1 and 2, below) and provides a brief outline of the physics parameters and power peaking analyses performed, including the application of uncertainty factors. These changes make the methods consistent with current NRC approved methods.

Reference 1 - "Duke Power Company Nuclear Design Methodology for Core Operating Limits of Westinghouse Reactors", DPC-NE-2011P-A, March 1990.
Reference 2 - "Multidimensional Reactor Transient's and Safety Analysis Physics Parameter Methodology", DPC-NE-3001P-A, November 1991.
Reference 3 - "FSAR Chapter 15 System Transient Analysis Methodology", DPC-NE-3002-A, Revision 3, SER Dated February 5, 1999.

Question 6. Section 9.1.5, first paragraph. Please provide clarification of this change and the technical justification for it. Please provide comparison between the old sentence and the new sentence.

Response
Original Sentence: "Calculations using EPRI-NODE are run at these power levels and nominal conditions to provide predicted power distributions for comparison."

Proposed Sentence: "Calculations are performed at these power levels and nominal conditions to provide predicted power distributions for comparison."

Specifically the words "Calculations using EPRI-NODE are run" were changed to "Calculations are performed". This change makes the description in this section valid when other NRC approved design methods are used (for example, SIMULATE).

APPENDIX C

Original Issue NRC SER

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

March 13, 1985

Docket Nos: 50-369, 50-370
and 50-413, 50-414

Mr. H. B. Tucker, Vice President
Nuclear Production Department
Duke Power Company
422 South Church Street
Charlotte, North Carolina 28242

Dear Mr. Tucker:

Subject: Topical Report on Physics Methodology for Reloads: McGuire and Catawba Nuclear Station

In response to your letter of July 18, 1984, with its supplemental information provided on November 30, and December 19, 1984, the NRC staff and its contractor, Brookhaven National Laboratory (BNL), have reviewed Duke Power Company Topical Report DPC-NF-2010, entitled "McGuire Nuclear Station/Catawba Nuclear Station Nuclear Physics Methodology for Reload Design," dated April 1984. This topical report is the first of a sequence of topical reports planned in regards to reload design at these stations. It describes the fuel, physics codes, fuel cycle design methods, and derivation of core physics parameters. It also presents statistical benchmarks which quantify reactivity and power distribution uncertainties.

Enclosed is our Safety Evaluation Report (SER) for this review. The SER notes in Section 3 that Section 6.3 and Chapter 7 of the Topical Report were excluded in our evaluation. Section 6.3 discusses the systematic application of safety related physics parameters for reload safety evaluation and, therefore, is outside the scope of the methodology described in the report. Chapter 7 discusses application of the physics methods to power peaking analysis and will be reviewed following a future submittal on three-dimensional power peaking analysis. Apart from these exclusions, we find that the methodology in the report, as modified by Duke's supplemental information, is acceptable for referencing in licensing actions involving nuclear physics calculations for reload design for the McGuire and Catawba Nuclear Stations.

We do not intend to repeat our review of the matters described in the report and found acceptable when the report appears as a reference in license applications, except to assure that the material presented is applicable to the specific plant involved. Our acceptance applies only to the matters described in the report.

In accordance with procedures established in NUREG-0390, it is requested that you publish the accepted version of this report within three months of receipt of this letter. The accepted version shall incorporate this letter and the enclosed evaluation between the title page and the abstract. The accepted version shall include an -A (designating accepted) following the report identification symbol.

Should our criteria or regulations change such that our conclusions as to the acceptability of the report are invalidated, you will be expected to revise and resubmit the report or submit justification for the continued effective applicability of the topical report.

Sincerely,

Cecil O. Thomas

Cecil O. Thomas, Chief
Standardization and Special Projects Bra
Division of Licensing

Enclosure:
As stated

cc: See next page

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ENCLOSURE

SAFETY EVALUATION REPORT

Report Title:	McGuire Nuclear Station/Catawba Nuclear Station Nuclear Physics Methodology for Reload Design
Report Number:	DPC-NF-2010
Report Date:	April 1984
Originating Organization:	Duke Power Company
Reviewed By:	Core Performance Branch, BNL, and Core Performance Branch, NRC

1. Introduction

This report describes the methodology adopted by Duke Power Company for the physics analysis of the McGuire and Catawba nuclear reactors. The physics analysis (also referred to as the nuclear design process in the topical report) is intended to determine the values of safety related parameters including those describing the core power distribution, reactivity worths and coefficients, and the reactor kinetics characteristics. These values of the physics parameters are then intended to serve as input to the reload safety analysis.

2. Summary of Report

In this methodology the main computational tools used for the physics analysis are the EPRI-ARMP code system[2] and the CASMO-2[3] code. The fuel performance codes COMETHE-IIIK[4] and TACO-2[4] are used for fuel performance analyses. CASMO-2, using a processed version of the ENDF/B-3 library in either 69 or 25 groups, and EPRI-CELL, using a 97-group library derived from ENDF/B-4, are used for cross section generation. Strong absorbers are modeled with CASMO-2, and equivalent diffusion theory parameters are generated by matching reaction rates calculated with CASMO-2 and PDQ07[5]. An assembly colorset PDQ07 model is used to generate k_∞ and M^2 data for the EPRI-NODE-P 3-D simulator, while a quarter core PDQ07 model is used for the calculation of x-y power distributions,

control bank worths, boron and xenon worths, and temperature coefficients. The NODE-P model is used for 3-D power distributions, ejected rod worths, differential rod worths, and xenon transient calculations.

The report describes the procedures used to calculate integral and differential control rod worths, shutdown margins, ejected and dropped rod worths, trip reactivity, critical boron concentrations, boron worth, xenon worth, reactivity coefficients, kinetics parameters, radial power peaking, and local power peaking. Measured parameters for the first cycles of McGuire Units 1 and 2, and Sequoyah Unit 1 have been compared with calculated values. Measured and calculated power distributions have been analyzed statistically and 95/95 Observed Nuclear Reliability Factors (ONRF) have been extracted.

3. Summary of Evaluation

The nuclear physics methodology described in Topical Report DPC-NF-2010 is the first part of a reload safety evaluation methodology to be submitted by the licensee, which is expected to also include fuel performance analysis, thermal-hydraulics analysis and transient and accident analysis. The licensee has indicated that this reload methodology will include Reload Safety Analysis Checklist (RSAC) comparisons which will be submitted first in collaboration with the fuel vendor, and later independently by the licensee. The licensee has also indicated that a 3-D Power Peaking Analysis will be submitted separately and, consequently, Sections 7.1, 7.2, 7.3, 7.4 and 7.4.1 will be reviewed after this analysis has been submitted. Although the application of the physics parameters has been briefly discussed in Section 6.3, the systematic application of safety related physics parameters for reload safety evaluation is outside the scope of the methodology described in the topical report and, consequently, has also been excluded from this review. The focus of the present evaluation has been on the adequacy of the methodology for calculating safety related physics parameters for use in reload safety analyses. The reload design methods are discussed in the following sections.

A. Nuclear Code System and Calculational Procedures

The Duke Power nuclear methodology is based on the well known and benchmarked EPRI-ARMP system, CASMO-2 and PDQ07 codes. Additionally, the use of a similar system of nuclear codes has been approved[6] by the NRC for use by Duke Power in the design of reload cores for the Oconee Nuclear Station.[7] The fuel performance codes COMETHE-IIIK and TACO-2, which are used for generating fuel properties related input data for the nuclear codes, are also well known and widely used in the industry. The cross section libraries used with EPRI-CELL and CASMO-2 have been derived from either the ENDF/B-3 or the ENDF/B-4 library, and contain a sufficiently detailed energy structure to enable an accurate determination of safety related physics parameters. EPRI recommended procedures[8] are followed in the use of the nuclear code system. A sufficient number of branch calculations are performed with the PDQ07 colorset model (both at beginning of life (BOL) and at selected burnup points, varying moderator and fuel temperature, soluble boron concentration, control rod insertion and xenon concentration) to allow proper determination of boron, xenon, Doppler and control rod worths and the relevant reactivity coefficients.[11] Sufficiently small steps are taken during the depletion calculations with the quarter core PDQ07 model to properly account for the effects of exposure. Measured values of critical boron concentrations, control rod worths, ejected rod worths, and isothermal temperature coefficients for Cycle 1 of both McGuire Unit 1 and Unit 2 have been compared with predictions. The measured critical boron concentrations are reproduced to within about 60 ppm with a standard deviation of about 15 ppm. Control rod bank worths are reproduced with a standard deviation of less than 8%. The isothermal temperature coefficients are reproduced to within about 5 pcm/°F, with a standard deviation of 1.87 pcm/°F. The quality of agreement between measured and predicted values of these physics parameters is acceptable provided the uncertainties are properly considered in the safety analysis.

B. Safety Related Parameters and Their Application

Calculation and application of the safety related physics parameters are described in chapter 6 of the report. A list of selected reload safety

related physics parameters is given in Table 6-1. It should be noted however, that parameters such as fuel temperature, fuel rod pressure, core DNB limits, fuel census data, maximum critical boron concentration, maximum shutdown boron concentration, which are used in the reload safety analyses of Westinghouse reactors[9], do not appear in Table 6-1. The criteria for evaluating the safety of a reload core design are not specified in sufficient detail. Duke Power should include this information in future topical reports.

C. Kinetics Parameters

Kinetics parameters are calculated using PDQ07 and the DELAY code. The calculated kinetics parameters include the six group delayed neutron fractions and effective yields, the total effective delayed neutron fraction, the prompt neutron generation time, and reactivity versus positive and negative doubling time. PDQ07 is used to obtain spatially averaged isotopic fission rates as a function of burnup, and DELAY is used to calculate kinetics parameters and to relate the reactor period to the inserted reactivity. The kinetics parameters are generated for both beginning of cycle (BOC) hot zero power (HZP) and hot full power (HFP) conditions with all rods out (ARO). A second set of delayed neutron parameters is generated for end of cycle (EOC).

The codes and methodology employed for the determination of these parameters have been previously reviewed and approved[10] by the staff.

D. Radial Local Power Peaking Analysis

A quadrant symmetric EPRI-NODE model is used to calculate nodal power distributions. A full core EPRI-NODE model is used to evaluate non-symmetric power distributions such as those encountered in the dropped rod configuration. The nodal powers are multiplied by the corresponding assembly radial local factor to yield the calculated total peaking factor:

$$F^C_Q = \text{Max}\,\{F^{Node}_{i\ell} \times RL_\ell\}\,, \qquad (1)$$

where RL_ℓ is the radial local factor for assembly ℓ, and $F^{Node}_{i\ell}$ is the nodal

power calculated at the axial location i for the assembly ℓ. The reliability factor for F_Q, F_Q^R, is calculated such that 95% of the calculated powers will be greater than the measured powers at a confidence level of 95%. Applying an additional multiplier, F_Q^E, to account for manufacturing tolerance, the total peaking factor, F_Q^T is defined as

$$F_Q^T = F_Q^R \times F_Q^E \times F_Q^C. \qquad (2)$$

Duke Power Company has presented comparisons between PDQ07 and CASMO-2 predictions of pin powers for 10 fuel assemblies at HFP, BOL, and no xenon conditions. In addition, measured pin powers in cold critical assemblies have been compared to PDQ07 predictions in two cases[7]. None of the measured or calculated lattices had any control rods inserted. On the basis of these results, Duke Power concluded that the PDQ07 prediction of the peak pin power is always conservative with respect to CASMO-2 calculations and to measurement; therefore, no uncertainty in the calculated radial local power is required. In response to a request for additional information, Duke Power has provided (1) results from two cold critical measurements that Duke Power made as prime contractor to DOE (Report DOE/ET/34212-41) and (2) a comprehensive solution to a standard problem recently developed at BNL to evaluate calculations of typical PWR fuel assemblies. The thorough and detailed nature of the solution, supplied in a relatively short period of time, is clearly an indication of the resources available to Duke Power in making physics calculations and their familiarity with the methods and procedures applicable in these analyses.

Comparison of EPRI-CELL/PDQ07 predictions of peak pin powers to measurements for the two criticals in the DOE study show that the EPRI-CELL/PDQ07 predictions of peak pin power are conservative by ≈1%. Duke Power believes that the overprediction of pin powers near the water holes is attributable to the use of Mixed Number Density (MND) thermal cross sections. It should be noted, however, that the use of MND cross sections does not necessarily lead to an overprediction of peak pin powers.[12] Comparison of the Duke Power solution to the standard problem with the benchmark solutions shows that at

BOL the Duke Power methods do indeed overpredict the peak pin power by just over 1%. However, the Duke Power methods underpredict the peak pin power by approximately 1% at 40,000 MWD/MTU with the "cross over" occurring smoothly at approximately 15,000 MWD/MTU. The Duke Power predictions are expected to have a similar exposure dependence relative to measurement. Any conservatism that might be present in the methodology used by Duke Power at BOL is not expected to persist at all exposures.

The basic methods used by Duke Power to calculate local radial peaking factors are in wide use, and the uncertainties associated with them have been published.[12,13] A review of the literature indicates that the appropriate uncertainty is a standard deviation of 2% between measured local radial power peaking factors and those calculated with a fine mesh diffusion theory code. In an amendment[14] to DPC-NF-2010 Duke Power has accounted for a 2% uncertainty in the calculation of the local peaking factor. The corresponding revised values of F_Q^R and $F_{\Delta H}^R$ are discussed in Section 3F.

E. Assembly Axial Power Analysis

The EPRI-NODE-P model with 12 axial nodes underpredicts the axial power peaking by an average of 2.2%. This deficiency of the model has been discussed with the licensee, who has noted that the agreement of model prediction to measurement is improved if (1) the number of axial nodes is increased from 12 to 18, and (2) the rodded M^2 is linearly adjusted according to the control fraction in the node. Despite the underprediction of the axial peaking using the EPRI-NODE-P model with 12 axial nodes, the total peaking factor F_Q^T (Equation 2) is not underestimated since the observed nuclear reliability factor (ONRF), F_Q^R, accounts for the bias between measurement and prediction.

While the 12 node model is acceptable, it is recommended that the Duke Power Company use the EPRI-NODE-P model with 18 axial nodes per assembly in all calculations. The enhanced accuracy of the model will improve the representation of non-uniform axial effects in the fuel assemblies.

F. Statistical Analysis

In deriving the calculational uncertainty of the models, the difference between measured and calculated power peaking factors has been assumed to be a normally distributed random variable. The D'Test has been applied to the difference distributions to establish their normality. The one-sided upper tolerance limit (OSUTL) on the difference variable, D, is

$$OSUTL(\overline{D}) = D+KxS(D), \qquad (3)$$

where $\overline{D}$ is the mean value of the difference variable, S(D) is the standard deviation, and K is the (sample size dependent) one-sided tolerance factor for the 95% probability at the 95% confidence level.

Using Equation (3), an upper limit to the calculated parameter can be defined as

$$UL(C) = \overline{M}-\overline{D}+KxS(D), \qquad (4)$$

where $\overline{M}$ is the mean of the measured variable. Finally, the observed nuclear reliability factor (ONRF) is defined

$$ONRF = UL(C)/\overline{M}. \qquad (5)$$

Utilizing 1038 observations (i.e., comparisons between measurements and predictions), the assembly peak axial ONRF (F_Q^R) has been determined by Duke Power to be 1.058, using the values; $\overline{M}$ = 1.375, $\overline{D}$ = 0.031, S (D) = 0.028 and K = 1.7259.

As noted in Section 3D, this value of F^R_Q assumes that there is no uncertainty in the calculation of the local power peaking factor. If, as indicated in reference 14, a fractional uncertainty of .02 is assumed for the local peaking, then by statistically combining the uncertainties for manufacturing tolerance (.03), assembly axial peaking (0.035), and local peaking (.02) the following

reliability factor for the total peaking, F_Q^{SCUF}, is obtained

$$F_Q^{SCUF} = 1 + (.031/1.375) + [(.03)^2 + (.035)^2 + (.02)^2]^{1/2} = 1.073. \qquad (6)$$

The corresponding Duke Power analysis for the radial ONRF ($F_{\Delta H}^R$) using $\bar{M}$ = 1.131, $\bar{D}$ = 0.002, S(D) = 0.02 and K = 1.7343 (846 Observations) results in an ONRF ($F_{\Delta H}^R$) of 1.029. As in the case of the F_Q^{SCUF}, combining the uncertainties due to manufacturing tolerance (.03), the radial assembly peaking (.03) and the radial local peaking (.02) yields

$$F_{\Delta H}^{SCUF} = 1 + [(.03)^2 + (.03)^2 + (.02)^2]^{1/2} = 1.047. \qquad (7)$$

These values for F_Q^{SCUF} and $F_{\Delta H}^{SCUF}$ include a 2% allowance for uncertainty in the calculation of the local peaking factor and are acceptable.

4. CONCULSION

The Duke Power Company Topical Report on Nuclear Physics Methodology for Reload Design (DPC-NR-2010) has been reviewed. As noted in Section 3 above, Sections 6.3, 7.1, 7.2, 7.3, 7.4, and 7.4.1 of the Topical Report were excluded from this evaluation.

Apart from these exclusions the methodology described in DPC-NF-2010 and modified in Reference 14 is found to be acceptable for referencing in licensing documents for the McGuire and Catawba Nuclear Stations.

References

1. "Advanced Recycle Methodology Program System Documentation," Nuclear Associates International, CCM-3, September 1977.

2. "CASMO-2 A Fuel Assembly Burnup Program," Studsvik Energiteknik AB, Studsvik /NP-81/3, 1981.

3. "COMETHE-IIIK-A Computer Code for Predicting Mechanical and Thermal Behavior of a Fuel Pin," Belgonuclear S.A., BN7609.1, March 1977.

4. "TACO2-Fuel Pin Performance Analysis," Babcock & Wilcox, BAW-10411, Lynchburg, Virginia, August 1979.

5. "PDQ-7 Reference Manual II," Bettis Atomic Power Laboratory, WAPD-TM-947(L), February 1971.

6. Letter from P. Wagner, NRC, to W. O. Parker, Jr., providing results of review of Duke Power Company Technical Report NSF-1001 on Oconee Nuclear Station Reload Design Methodology, July 29, 1981.

7. "Duke Power Company/Oconee Nuclear Station/Reload Design Methodology" Technical Report NFS-1001," Duke Power Company, NFS-1001, Rev. 1, May 1980.

8. B. M. Rothleder, et al., "PWR Core Modeling Procedures for Advanced Recycle Methodology Program," RP-976-1, August 1979.

9. "Westinghouse Reload Safety Evaluation Methodology (WCAP-9272)," F. M. Boredelon et al., Westinghouse Electric Corporation, March 1978.

10. Letter, W. O. Parker to H. R. Denton, "Oconee Reload Design Methodology Topical Report," Question 3, Docket Nos. 50-269, 270, 287, November 13, 1980.

11. Letter, H. B. Tucker (Duke Power) to H. R. Denton (NRC), "Response Request for Additional Information Regarding Topical Report DPC-NF-2010," Docket Nos., 50-369/370, 50-413/414, November 30, 1984.

12. R. D. Mosteller and R. S. Borland, "COPHIN Code Description," EPRI NP-1385, Electric Power Research Institute, April 1980.

13. P. L. Langford Jr., and R. J. Nath, "Evaluation of Nuclear Hot Channel Factor Uncertainties," WCAP-7398-L, Westinghouse Electric Corporation, April 1969.

14. Letter, H. B. Tucker (Duke Power) to H. R. Denton (NRC), "Response Request for Additional Information Regarding Topical Report DPC-NF-2010, "Docket Nos., 50-369/370, 50-413/414, December 19, 1984.

APPENDIX D

Revision 1 NRC SER

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

October 1, 2002

Mr. H. B. Barron
Vice President, McGuire Site
Duke Energy Corporation
12700 Hagers Ferry Road
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SUBJECT: McGUIRE NUCLEAR STATION, UNITS 1AND 2 RE: ISSUANCE OF AMENDMENTS (TAC NOS. MB3222 AND MB3223)

Dear Mr. Barron:

The Nuclear Regulatory Commission has issued the enclosed Amendment No. 208 to Facility Operating License NPF-9 and Amendment No. 189 to Facility Operating License NPF-17 for the McGuire Nuclear Station, Units 1 and 2. The amendments consist of changes to the Technical Specifications in response to your application dated October 7, 2001, as supplemented by letter dated August 7, 2002.

The amendments revise TS 5.6.5.a by adding a few parameter limits currently included in the Core Operating Limits Report. In addition to the license amendment request, you also submitted revisions to four previously approved topical reports for the Nuclear Regulatory Commission staff review and approval. The enclosed Safety Evaluation also addresses these topical reports.

A Notice of Issuance will be included in the Commission's biweekly *Federal Register* notice.

Sincerely,

Robert Martin

Robert E. Martin, Senior Project Manager, Section 1
Project Directorate II
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket Nos. 50-369 and 50-370

Enclosures:
1. Amendment No. 208 to NPF-9
2. Amendment No. 189 to NPF-17
3. Safety Evaluation

cc w/encls: See next page

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UNITED STATES
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SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 208 TO FACILITY OPERATING LICENSE NPF-9

AND AMENDMENT NO. 189 TO FACILITY OPERATING LICENSE NPF-17

DUKE ENERGY CORPORATION

MCGUIRE NUCLEAR STATION, UNITS 1 AND 2

DOCKET NOS. 50-369 AND 50-370

1.0 INTRODUCTION

By letter dated October 7, 2001, as supplemented by letter dated August 7, 2002, Duke Power Company, et al. (DPC, the licensee), submitted a request for changes to the McGuire Nuclear Station, Units 1 and 2, Technical Specifications (TS).

Revisions were proposed for TS 5.6.5.a, Item 1, to add the moderator temperature coefficient (MTC) 60 parts per million (ppm) surveillance limit. The specific value of the surveillance limit was previously relocated to the Core Operating Limits Report (COLR). A new item 12, "31 EFPD surveillance penalty factors for Specifications 3.2.1 and 3.2.2," is also proposed to be added to TS 5.6.5.a.

The initial submittal, dated October 7, 2001, proposed to change the dates and revision numbers for three of the Nuclear Regulatory Commission (NRC) approved analytical methods previously listed in TS 5.6.5.b, as listed below. The changes would reflect later versions of these topical reports that were also submitted with the October 7, 2001, submittal for NRC review and approval. As required by TS 5.6.5.b, only those methods listed within the TS as having been reviewed and approved by the NRC, can be used to determine the subject core operating limits. The subject core operating limits are listed in TS 5.6.5.a and their values are located in the COLR. A revision to a fourth report, DPC-NE-1003, was also submitted for NRC review and approval.

- DPC-NE-2009, Revision 1, "Duke Power Company Westinghouse Fuel Transition Report," August 2001.
- DPC-NF-2010, Revision 1, "Duke Power Company McGuire Nuclear Station and Catawba Nuclear Station Nuclear Physics Methodology for Reload Design," August 2001.
- DPC-NE-2011, Revision 1, "Duke Power Company Nuclear Design Methodology Report for Core Operating Limits of Westinghouse Reactors," August 2001.
- DPC-NE-1003, Revision 1, "McGuire Nuclear Station and Catawba Nuclear Station Rod Swap Methodology Report for Startup Physics Testing," August 2001.

The licensee in its letter of October 7, 2001, stated that, once approved, the approved topical report revisions, except for DPC-1003, Revision 1, will be listed in Section 5.6.5.b of the McGuire TS, to replace their respective original versions, and that the approved version of DPC-NE-2011-P, Revision 1, will also be listed in the references for TS Bases 3.2.1 and 3.2.3 to replace the existing reference to the original version, DPC-NE-2011-P-A.

However, on July 10, 2002, the NRC issued amendments numbered 203 and 184 to the McGuire Unit 1 and 2 operating licenses that effectively relocated the topical report revision numbers and dates from the TS 5.6.5.b list of approved methodologies to the COLR. Amendments 203 and 184 were consistent with the NRC Technical Specification Task Force (TSTF) Standard TS Traveler TSTF-363, "Revise Topical Report References in ITS 5.6.5 COLR." Accordingly, since this portion of its request is no longer needed in view of amendments 203 and 184, the licensee's letter dated August 7, 2002, eliminated the requests to change TS 5.6.5.b and proposed revisions to BASES 3.2.1 and 3.2.3 to make its submittal consistent with the implementation of amendments 203 and 184 at the McGuire Nuclear Station. Nonetheless, this Safety Evaluation sets forth the NRC staff's evaluation of the licensee's proposed changes to the topical reports listed above.

2.0 BACKGROUND

Title 10 of the *Code of Federal Regulations* (10 CFR) Section 50.36 (c)(2)(ii)(B), Criterion 2, specifies that a process variable, design feature, or operating restriction that is an initial condition of a design basis accident or transient analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier must be included in the TS limiting conditions for operation (LCO). Accordingly, the reactor operating parameters, which are the initial conditions for the safety analyses of the design basis transients and accidents, are included in the TS LCOs.

Since many parameter limits, such as core physics parameters, generally change with each reload core, licensees previously needed to request TS amendments to update these parameters for each refueling cycle. NRC Generic Letter (GL) 88-16 (Ref. 4) provides guidance for relocating the values of the cycle-specific core operating parameter limits from TS to the COLR, thus eliminating unnecessary burden on the licensees and the NRC to update these limits in the TS for each fuel cycle. The guidance includes adding the COLR in the TS administrative reporting requirement that also specifies (1) the cycle-specific parameters included in the COLR, and (2) the analytical methods that the NRC has previously reviewed and approved to be used to determine the core operating parameters limits.

The McGuire TS 5.6.5, "Core Operating Limits Report (COLR)," conforms to GL 88-16 guidance. TS 5.6.5.a lists a set of parameters, including the reference to the actual TS number for each specified parameter. TS 5.6.5.b specifies the topical reports that are used for the determination of the core operating limits.

The proposed TS changes in this license amendment request are to revise the parameters listed in TS 5.6.5.a. These revisions are based on the guidance of GL 88-16.

3.0 STAFF EVALUATION

In this section, the staff will discuss the review of the revised versions of the four previously approved topical reports submitted for staff review, and the proposed TS changes.

3.1 Topical Reports Revisions

The licensee requested the NRC to review revisions to four topical reports that were previously approved and listed in TS 5.6.5.b as the approved methodologies used for the determination of the parameter limits in the COLR. Since the staff has reviewed and approved the original versions of these topical reports, the staff review of these revised versions concentrated on the revisions made to the approved reports.

3.1.1 DPC-NE-2009, Revision 1

Topical report, DPC-NE-2009-P-A, (Ref. 5), provides general information about the Robust Fuel Assembly (RFA) design and describes methodologies used for reload design analyses to support the licensing basis for use of RFAs in the McGuire and Catawba reload cores. These methodologies include fuel rod mechanical reload analysis methodology and the core design, thermal-hydraulic analysis, and accident analysis methodologies. The NRC approved the report in September 1999.

Revision 1 of DPC-NE-2009, as amended by the August 7, 2002, letter (Ref. 2), consists of the following minor changes to its Chapter 6, "UFSAR Accident Analyses."

(A) Update of the reference list in Section 6.7 as follows:

- Update reference 6-25, WCAP-10054-P-A Addendum 2, to Revision 1, dated July 1997.
- Correct reference 6-35, WCAP-8354, with proprietary topical report number, and designate the second report as a non-proprietary report.
- Add reference 6-39, Westinghouse letter NSD-NRC-99-5839, "1998 Annual Notification of Changes to the Westinghouse Small Break LOCA and Large Break LOCA ECCS Evaluation Models, Pursuant to 10 CFR 50.46(a)(3)(ii)," dated July 15, 1999 (Ref. 6).

(B) Addition of a paragraph to Section 6.5.1, "Small Break LOCA," to explain that the Westinghouse small break LOCA NOTRUMP Evaluation Model includes the error corrections and model enhancements described in a few Westinghouse annual notifications required by 10 CFR 50.46, including the 1998 annual notification referenced in Reference 39.

The first two changes in the reference list are editorial and merely provide the latest version of the approved topical report or identify the proprietary and non-proprietary versions of a topical report. Reference 6-39, Westinghouse letter NSD-NRC-99-5839, is the annual notification of the changes to the LOCA evaluation models during 1998. This notification documented the following error corrections or model enhancements to the NOTRUMP small break LOCA Evaluation Model.

- A programming error correction on the SBLOCTA rod-to-rod radiation model, that is not modeled in licensing basis analyses and therefore, has no impact on the small break LOCA results.

- A logic simplification to the NOTRUMP droplet fall model that produces insignificant differences in results.

- A change in the reactor coolant pump heat in NOTRUMP that is not used in the evaluation model and therefore, has no impact on the small break LOCA results.

- A modification of NOTRUMP steam generator tube condensation heat transfer logic for a foreign plant that does not affect standard Westinghouse Pressurized Water Reactor calculations.

- An extension of reactor coolant conditions to allow for the NOTRUMP point kinetics calculations to be performed for cases that experience core uncovery conditions prior to reactor trip. For typical small break LOCA analyses, the reactor trips long before any threat of core uncovery and therefore, the change has no impact on peak cladding temperature calculations.

- A programming change in SBLOCTA code to allow for modeling of variable length blankets on either ends of the rod that involves no changes to the thermal-hydraulic fuel rod model, nor the solution technique.

Since the changes documented in the Westinghouse annual notice have insignificant impact on the small break LOCA analyses, the staff concludes the addition of Reference 6-39 is acceptable. Therefore, Revision 1 of DPC-NE-2009-P-A, as modified in the August 7, 2002, letter, is acceptable.

3.1.2 DPC-NF-2010, Revision 1

Topical Report DPC-NF-2010, (Ref. 7), describes DPC's Nuclear Design Methodology for McGuire and Catawba Nuclear Stations. The nuclear design process consists of mechanical properties used as nuclear design input, the nuclear code system and methodology that DPC intends to use to perform design calculations and to provide operational support, and the development of statistical factors.

Revision 1 of DPC-NF-2010, updates the report to permit the use of certain methods approved subsequent to the implementation of the original version, such as the use of CASMO-3/ SIMULATE-3P reactor physics methods (Ref. 8). Other changes are made to reflect revisions to the core design parameters such as shutdown margin, boron and control rod worth, axial and radial peaking factors, and cycle length, as well as numerous editorial changes.

During the review, the staff also identified a few discrepancies associated with administrative changes. In response to the staff's request for additional information (Ref. 2), the licensee provided further changes to Revision 1 of the topical report. These modifications include clarifications to revised sections and minor changes to equations. The NRC staff has reviewed the analyses associated with the changes to Topical Report DPC-NF-2010 and the responses to the requests for additional information pertaining to these changes. The staff has concluded

that the changes to this topical report consist mostly of administrative changes and clarifications to the original NRC approved topical report and that there are no unreviewed methodology or regulatory issues. Therefore, the staff finds the changes to be acceptable.

3.1.3 DPC-NE-2011, Revision 1

Topical Report DPC-NF-2011, (Ref. 9), describes the methodology for performing a maneuvering analysis for four-loop plants, such as the McGuire and Catawba Nuclear Stations. The licensee has developed this methodology as an alternate to the existing Relaxed Axial Offset Control (RAOC) Methodology. The licensee pointed out that this maneuvering analysis results in several advantages: more flexible and prompt engineering support for the operating stations, consistency with the methods of the licensee's nuclear design process, and potential increases in available margin through the use of three-dimensional monitoring techniques. The increase in margin occurs in limits on power distribution, control rod insertion, and power distribution inputs to the overpower delta-temperature and over-temperature delta-temperature reactor protection system (RPS) trip functions.

Revision 1 of DPC-NE-2011, updates the report to include editorial changes, and to permit the use of certain methods approved subsequent to the implementation of the original version, such as the CASMO-3/SIMULATE-3P methodology (Ref. 8). Other changes are made to reflect revisions to the core design parameters such as power peaking factors, axial and radial power distributions, and cycle length, as well as numerous editorial changes.

In response to the NRC staff's request for additional information (Ref. 2), the licensee provided additional information regarding cycle depletion times to clarify issues associated with power peaking versus burnup as a function of cycle time. The licensee's amendment request also included clarifications to revised sections and minor changes to equations. The NRC staff has reviewed the analyses associated with the changes to Topical Report DPC-NE-2011-A and the responses to the requests for additional information pertaining to the requested changes. Since the changes to this topical report consist mostly of administrative changes and clarifications to the original NRC approved topical report, the staff finds the changes to be acceptable.

3.1.4 DPC-NE-1003, Revision 1

Topical Report DPC-NE-1003 (Ref. 10), describes the measurement procedure used to determine the inferred bank worth and the calculation procedures used to develop the rod swap correction factor that accounts for the effect of a test bank on the partial integral worth of the reference bank. The NRC approved the report in May 1987 (Ref. 11) for rod worth measurement of reload cores for McGuire and Catawba Stations, Units 1 and 2.

Revision 1 of DPC-NE-1003 updates the report to permit the use of certain methods approved subsequent to the implementation of the original version, such as the use of CASMO-3/ SIMULATE-3P reactor physics methods (Ref. 8). Other changes are made to reflect the revision of the rod swap measurement procedures, and various editorial changes. In response to staff questions, the licensee, in its letter of August 7, 2002, provided the current version of the control rod worth measurement rod swap procedures, PT/0/A/4150/11A, dated January 19, 1996. The staff review of this current control rod worth measurement procedure has found it to be acceptable. The licensee, in the August 7, 2002, letter also modified the equation in Section 3 of the topical report for the calculation of the inferred rod bank worth from the

measured reference bank worth and bank height. This change is consistent with the equation described in step 12.12.5 of the current measurement procedures of January 19, 1996. Therefore, Revision 1 of DPC-NE-1003, as modified in the August 7, 2002, letter, is acceptable.

3.2 Proposed TS Changes

This section addresses the staff's evaluation of the proposed changes to TS 5.6.5.a regarding the cycle-specific operating parameters specified in the COLR. The staff review of these TS changes are based on the guidance of GL 88-16.

TS 5.6.5.a provides a list of core operating limits that are established prior to each reload cycle, or prior to any remaining portion of a reload cycle. The values of the limits are located in the COLR. For McGuire Nuclear Station, Units 1 and 2, the licensee proposed to revise the list by:

(1) adding "60 ppm" to Item 5.6.5.a.1 regarding the moderator temperature coefficient (MTC) surveillance limit for Specification 3.1.3, and

(2) adding Item 5.6.5.a.12, "31 EFPD surveillance penalty factors for Specifications 3.2.1 and 3.2.2."

These changes are evaluated below.

3.2.1 MTC 60 ppm Surveillance Limit

McGuire TS LCO 3.1.3 specifies that the MTC be maintained within the LCO limits, which are based on the safety analysis assumptions. For verification that these LCO limits are met, the Surveillance Requirements of TS 3.1.3 also place surveillance limits for conducting the end of cycle MTC measurement at boron concentrations of 300 ppm and 60 ppm. The LCO limits and the 300 ppm and 60 ppm surveillance limits are specified in the COLR. However, TS Item 5.6.5.a.1 operating limits does not currently identify the 60-ppm surveillance limit.

The proposed change to the McGuire TS would add the 60 ppm surveillance limit in Item 5.6.5.a.1. The new TS would read "Moderator Temperature Coefficients BOL and EOL limits and 60 ppm and 300 ppm surveillance limit for Specification 3.1.3." The NRC approved incorporating the 60-ppm surveillance limits into the COLR during the Improved Technical Specifications conversion in 1998 (Ref. 12 and 13); however, reference to this surveillance was not included in TS Item 5.6.5.a.1 at that time. The proposed TS change to include the 60 ppm surveillance limit in TS Item 5.6.5.a.1 provides consistency with previously approved requirements and, therefore, it is acceptable.

3.2.2 Relocation of Hot Channel Factors Surveillance Penalty Factors to COLR

Surveillance Requirements in TS 3.2.1 and 3.2.2, respectively, require that the heat flux hot channel factor, F_q (x,y,z), and the enthalpy rise hot channel factor, $F_{\Delta h}$ (x,y), be measured every 31 effective full power days (EFPD) during equilibrium conditions using the incore detector system to verify they are within the respective limits. To address the possibility that these hot channel factors may increase and exceed their allowable limits between surveillances, penalty factors are applied to these hot channel factors if their margins to the respective limits have decreased since the previous surveillance. These margin-decrease penalty factors are

calculated by projecting the limiting hot channel factors over the 31 EFPD surveillance intervals with the maximum changes at the limiting core location, and are based on reload core design. In Section 8, "Improved Technical Specification Changes," of DPC-NE-2009, the licensee proposed to replace the penalty factors with tables of penalty value as a function of burnup in the COLR to facilitate cycle-specific updates. TS Item 5.6.5.b.14 lists topical report DPC-NE-2009-P-A that includes (in response to a staff question during the review of DPC-NE-2009) the approved methodology used to calculate these burnup-dependent penalty factors. The staff found the methodology and the inclusion of the burnup-dependent margin decrease penalty factors in the COLR acceptable, as stated in the staff's Safety Evaluation supporting license Amendment Nos. 188 and 169, respectively, for McGuire Nuclear Station, Units 1 and 2 (Ref. 14).

The proposed changes to the McGuire TS would add Item 5.6.5.a.12 that reads: "31 EFPD surveillance penalty factors for Specifications 3.2.1 and 3.2.2." The addition of TS Item 5.6.5.a.12 would make it consistent with the previous staff approval of including these surveillance penalty factors in the COLR and, therefore, this proposed change is acceptable.

4.0 SUMMARY

The staff has reviewed the revisions to four previously approved topical reports described in Section 1.0 of this Safety Evaluation, and the proposed changes to McGuire Nuclear Station, Units 1 and 2, TS 5.6.5.a related to the COLR. Based on our evaluation, described in Section 3 of this Safety Evaluation, the staff concludes that the these topical report revisions, as amended by the August 7, 2002, letter, and the TS changes are acceptable.

5.0 STATE CONSULTATION

In accordance with the Commission's regulations, the North Carolina State official was notified of the proposed issuance of the amendments. The State official had no comments.

6.0 ENVIRONMENTAL CONSIDERATION

The amendments change recordkeeping, reporting, or administrative procedure 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 Commission has previously issued a proposed finding that the amendments involve no significant hazards consideration, and there has been no public comment on such finding (67FR 54680). 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.

7.0 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.

8.0 REFERENCES

1. Letter from M. S. Tuckman, Duke Energy Corporation, to US Nuclear Regulatory Commission, "Duke Energy Corporation; Catawba Nuclear Station Units 1 and 2, Docket Nos. 50-413, 50-414; McGuire Nuclear Station Units 1 and 2, Docket Nos. 50-369, 50-370; License Amendment Request Applicable to Technical Specifications 5.6.5, Core Operating Limits Report; Revisions to BASES 3.2.1 and 3.2.3; and Revisions to Topical Reports DPC-NE-2009-P, DPC-NF-2010, DPC-NE-2011-P, and DPC-NE-1003," October 7, 2001.

2. Letter from M. S. Tuckman, Duke Energy Corporation, to US Nuclear Regulatory Commission, "Duke Energy Corporation; McGuire Nuclear Station Units 1 and 2, Docket Nos. 50-369 and 370; Catawba Nuclear Station Units 1 and 2, Docket Nos. 50-413 and 414; Response to NRC Request for Additional Information - TAC nos. MB3222, MB3223, MB3343 and MB3344) and License Amendment Request Supplement," August 7, 2002.

3. Letter from M. S. Tuckman, Duke Energy Corporation, to US Nuclear Regulatory Commission, "License Amendment Request Applicable to the Technical Specifications Requirements for the Core Operating Limits Report - Oconee, McGuire, and Catawba Technical Specifications 5.6.5," December 20, 2001.

4. Letter from Dennis Crutchfield, USNRC, to All Power Reactor Licensees and Applicants, "Removal of Cycle-Specific Parameter Limits from Technical Specifications (Generic Letter 88-16)," October 4, 1988.

5. DPC-NE-2009-P-A, "Duke Power Company Westinghouse Fuel Transition Report," December 1999.

6. Letter from J. S. Galembush, Westinghouse Electric Company, to US Nuclear Regulatory Commission, "1998 Annual Notification of Changes to the Westinghouse Small Break LOCA and Large Break LOCA ECCS Evaluation Models, Pursuant to 10 CFR 50.46(a)(3)(ii)," NSD-NRC-99-5839, July 15, 1999.

7. DPC-NF-2010A, "Duke Power Company McGuire Nuclear Station and Catawba Nuclear Station Nuclear Physics Methodology for Reload Design," June 1985..

8. DPC-NE-1004A, Revision 1, "Nuclear Design Methodology Using CASMO-3/ SIMULATE-3P," SER dated April 26, 1997.

9. DPC-NE-2011, "Nuclear Design Methodology Report for Core Operating Limits of Westinghouse Reactors," March 1990.

10. DPC-NE-1003, "Rod Swap Methodology Report for Startup Physics Testing," December 1986.

11. Letter from Darl Hood, USNRC, to H. B. Tucker, Duke Power Company, "Rod Swap Methodology Report for Startup Physics Testing, McGuire and Catawba Nuclear Stations, Units 1 and 2 (TACs 62981, 62982, 62983, 62984)," May 22, 1987.

12. Letter from Frank Rinaldi, USNRC, to H. B. Brown, McGuire Site, Duke Energy Corporation, "Issuance of Amendments - McGuire Nuclear Station, Units 1 and 2, (TAC Nos. M98964 and M98965)," September 30, 1998.

13. Letter from Peter Tam, USNRC, to G. R. Peterson, Catawba Nuclear Station, Duke Energy Corporation, "Issuance of Amendments - Catawba Nuclear Station, Units 1 and 2 (TAC Nos. M95298 and M95299)," September 30, 1998.

14. Letter from Frank Rinaldi, USNRC, to H. B. Brown, McGuire Site, Duke Energy Corporation, "McGuire Nuclear Station, Units 1 and 2, Re: Issuance of Amendments (TAC Nos. MA2411 and MA2412)," September 22, 1999.

15. Letter from Peter Tam, USNRC, to G. R. Peterson, Catawba Nuclear Station, Duke Energy Corporation, "Catawba Nuclear Station, Units 1 and 2, Re: Issuance of Amendments (TAC Nos. MA2359 and MA2361)," September 22, 1999.

16. Letter from Robert F. Martin, USNRC, to David L. Rehn, Catawba Site, Duke Power Company, "Issuance of Amendments - Catawba Nuclear Station, Units 1 and 2 Cycle Specific Parameters to the Core Operating Limits Report (TAC Nos. M85472 and M85473)," March 25, 1994.

Principal Contributor: Y. Hsii
A. Attard

Date: October 1, 2002

EC050

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

October 1, 2002

RECEIVED
OCT - 8 2002
REGULATORY COMPLIANCE

Mr. G. R. Peterson
Site Vice President
Catawba Nuclear Station
Duke Energy Corporation
4800 Concord Road
York, South Carolina 29745-9635

SUBJECT: CATAWBA NUCLEAR STATION, UNITS 1 AND 2 RE: ISSUANCE OF AMENDMENTS (TAC NOS. MB3343 AND MB3344)

Dear Mr. Peterson:

The Nuclear Regulatory Commission has issued the enclosed Amendment No. 202 to Facility Operating License NPF-35 and Amendment No.195 to Facility Operating License NPF-52 for the Catawba Nuclear Station, Units 1 and 2. The amendments consist of changes to the Technical Specifications (TS) in response to your application dated October 7, 2001, as supplemented by letter dated August 7, 2002.

The amendments revise TS 5.6.5.a by adding a few parameter limits currently included in the Core Operating Limits Report. In addition to the license amendment request, you also submitted revisions to four previously approved topical reports for the Nuclear Regulatory Commission staff review and approval. The enclosed Safety Evaluation also address these topical reports.

A Notice of Issuance will be included in the Commission's biweekly Federal Register notice.

Sincerely,

Chandu P. Patel

Chandu P. Patel, Project Manager, Section 1
Project Directorate II /RA/
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket Nos. 50-413 and 50-414

Enclosures:
1. Amendment No. 202 to NPF-35
2. Amendment No. 195 to NPF-52
3. Safety Evaluation

cc w/encls: See next page

Catawba Nuclear Station

cc:

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 202 TO FACILITY OPERATING LICENSE NPF-35

AND AMENDMENT NO. 195 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

1.0 INTRODUCTION

By letter dated October 7, 2001, as supplemented by letter dated August 7, 2002, Duke Energy Corporation, et al. (DEC, the licensee), submitted a request for changes to the Catawba Nuclear Station, Units 1 and 2, Technical Specifications (TS).

Revisions were proposed for TS 5.6.5.a, Item 1, to add the moderator temperature coefficient (MTC) 60 parts per million (ppm) surveillance limit. The specific value of the surveillance limit was previously relocated to the Core Operating Limits Report (COLR). Two new items were also proposed to be added to TS 5.6.5.a. These two items are (1) Item 12, "31 EFPD surveillance penalty factors for Specifications 3.2.1 and 3.2.2," and (2) Item 13, "Reactor makeup water pumps combined flow rates limit for Specifications 3.3.9 and 3.9.2."

The initial submittal, dated October 7, 2001, proposed to change the dates and revision numbers for three of the Nuclear Regulatory Commission (NRC) approved analytical methods previously listed in TS 5.6.5.b, as listed below. The changes would reflect later versions of these topical reports that were also submitted with the October 7, 2001, submittal for NRC review and approval. As required by TS 5.6.5.b, only those methods listed within the TS as having been reviewed and approved by the NRC, can be used to determine the subject core operating limits. The subject core operating limits are listed in TS 5.6.5.a and their values are located in the COLR. A revision to a fourth report, DPC-NE-1003, was also submitted for NRC review and approval.

- DPC-NE-2009, Revision 1, "Duke Power Company Westinghouse Fuel Transition Report," August 2001.
- DPC-NF-2010, Revision 1, "Duke Power Company McGuire Nuclear Station and Catawba Nuclear Station Nuclear Physics Methodology for Reload Design," August 2001.
- DPC-NE-2011, Revision 1, "Duke Power Company Nuclear Design Methodology Report for Core Operating Limits of Westinghouse Reactors," August 2001.

- DPC-NE-1003, Revision 1, "McGuire Nuclear Station and Catawba Nuclear Station Rod Swap Methodology Report for Startup Physics Testing," August 2001.

The licensee in its letter of October 7, 2001, stated that, once approved, the approved topical report revisions, except for DPC-1003, Revision 1, will be listed in Section 5.6.5.b of the Catawba TS, to replace their respective original versions, and that the approved version of DPC-NE-2011-P, Revision 1, will also be listed in the references for TS Bases 3.2.1 and 3.2.3 to replace the existing reference to the original version, DPC-NE-2011-P-A.

However, on July 2, 2002, the NRC issued amendments numbered 199 and 192 to the Catawba Unit 1 and 2 operating licenses that effectively relocated the topical report revision numbers and dates from the TS 5.6.5.b list of approved methodologies to the COLR. Amendments 199 and 192 were consistent with the NRC Technical Specification Task Force (TSTF) Standard TS Traveler TSTF-363, "Revise Topical Report References in ITS 5.6.5 COLR." Accordingly, since this portion of its request is no longer needed in view of amendments 199 and 192, the licensee's letter dated August 7, 2002, eliminated the requests to change TS 5.6.5.b and proposed revisions to BASES 3.2.1 and 3.2.3 to make its submittal consistent with the implementation of amendments 199 and 192 at the Catawba Nuclear Station. Nonetheless, this Safety Evaluation sets forth the NRC staff's evaluation of the licensee's proposed changes to the topical reports listed above.

2.0 BACKGROUND

Title 10 of the *Code of Federal Regulation* (10 CFR) Section 50.36 (c)(2)(ii)(B), Criterion 2 specifies that a process variable, design feature, or operating restriction that is an initial condition of a design basis accident or transient analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier must be included in the TS limiting conditions for operation (LCO). Accordingly, the reactor operating parameters, which are the initial conditions for the safety analyses of the design basis transients and accidents, are included in the TS LCO.

Since many parameters limits, such as core physics parameters, generally change with each reload core, licensees need to request TS amendments to update these parameters for each refueling cycle. NRC Generic Letter (GL) 88-16 (Ref. 4) provides guidance for relocating the values of the cycle-specific core operating parameter limits from TS to the COLR, and thus eliminates the unnecessary burden on the licensees and the NRC to update these limits in the TS each fuel cycle. The guidance includes adding the COLR in the TS administrative reporting requirement that also specifies (1) the cycle-specific parameters included in the COLR, and (2) the analytical methods that the NRC has previously reviewed and approved to be used to determine the core operating parameters limits.

The Catawba TS 5.6.5, "Core Operating Limits Report (COLR)," conforms to the GL 88-16 guidance. TS 5.6.5.a lists a set of parameters, including the reference to the actual TS number for each specified parameter. TS 5.6.5.b specifies the topical reports that are used for the determination of the core operating limits.

The proposed TS changes in this license amendment request are to revise the parameters listed in TS 5.6.5.a. These revisions are based on the guidance of GL 88-16.

3.0 STAFF EVALUATION

In this section, the staff will discuss the review of the revised versions of the four previously approved topical reports submitted for staff review, and the proposed TS changes.

3.1 Topical Reports Revisions

The licensee requested the NRC to review revisions of four topical reports that were previously approved and listed in TS 5.6.5.b as the approved methodologies used for the determination of the parameter limits in the COLR. Since the staff has reviewed and approved the original versions of these topical reports, the staff review of these revised versions will concentrate on the revisions made to the approved reports.

3.1.1 DPC-NE-2009, Revision 1

Topical report, DPC-NE-2009-P-A, (Ref. 5), provides general information about the Robust Fuel Assembly (RFA) design and describes methodologies used for reload design analyses to support the licensing basis for use of the RFA design in the McGuire and Catawba reload cores. These methodologies include fuel rod mechanical reload analysis methodology and the core design, thermal-hydraulic analysis, and accident analysis methodologies. The NRC approved the report in September 1999.

Revision 1 of DPC-NE-2009-A, as amended by the August 7, 2002, letter (Ref. 2), consists of the following minor changes to Chapter 6, "UFSAR Accident Analyses:"

(A) Update of the reference list in Section 6.7 as follows:

- Update reference 6-25, WCAP-10054-P-A Addendum 2, to Revision 1, dated July 1997.
- Correct reference 6-35, WCAP-8354, with proprietary topical report number, and designate the second report as a non-proprietary report.
- Add reference 6-39 a Westinghouse letter NSD-NRC-99-5839, "1998 Annual Notification of Changes to the Westinghouse Small Break LOCA and Large Break LOCA ECCS Evaluation Models, Pursuant to 10 CFR 50.46(a)(3)(ii)," dated July 15, 1999 (Ref. 6).

(B) Addition of a paragraph to Section 6.5.1, "Small Break LOCA," to explain that the Westinghouse small break LOCA NOTRUMP Evaluation Model includes the error corrections and model enhancements described in a few Westinghouse annual notifications required by 10 CFR 50.46, including the 1998 annual notification referenced in Reference 39.

The first two changes in the reference list are editorial and merely provide the latest version of the approved topical report or identify the proprietary and non-proprietary versions of a topical report. Reference 6-39, the Westinghouse letter NSD-NRC-99-5839, is the annual notification of the changes to the LOCA evaluation models during 1998. This notification documented the following error corrections or model enhancements to the NOTRUMP small break LOCA Evaluation Model:

- A programming error correction on the SBLOCTA rod-to-rod radiation model that is not modeled in licensing basis analyses and therefore, has no impact on the small break LOCA results.

- A logic simplification to the NOTRUMP droplet fall model that produces insignificant differences in results.

- A change in the reactor coolant pump heat in NOTRUMP that is not used in the evaluation model and therefore, has no impact on the small break LOCA results.

- A modification of NOTRUMP steam generator tube condensation heat transfer logic to a foreign plant that does not affect standard Westinghouse Pressurized Water Reactor calculations.

- An extension of reactor coolant conditions to allow for the NOTRUMP point kinetics calculations to be performed for cases that experience core uncovery conditions prior to reactor trip. For typical small break LOCA analyses, the reactor trips long before any threat of core uncovery and therefore, the change has no impact on peak cladding temperature calculations.

- A programming change in SBLOCTA code to allow for modeling of variable length blankets on either ends of the rod that involves no changes to the thermal-hydraulic fuel rod model, nor the solution technique.

Since the changes documented in the Westinghouse annual notice have insignificant impact on the small break LOCA analyses, the staff concludes the addition of Reference 6-39 is acceptable. Therefore, Revision 1 of DPC-NE-2009-P-A, as modified in the August 7, 2002, letter, is acceptable.

3.1.2 DPC-NF-2010A, Revision 1

Topical Report DPC-NF-2010A, (Ref. 7), describes Duke Power Company's Nuclear Design Methodology for McGuire and Catawba Nuclear Stations. The nuclear design process consists of mechanical properties used as nuclear design input, the nuclear code system and methodology the licensee intends to use to perform design calculations and to provide operational support, and the development of statistical factors.

Revision 1 of DPC-NF-2010A, updates the report to permit the use of certain methods approved subsequent to the implementation of the original version, such as the use of CASMO-3/SIMULATE-3P reactor physics methods (Ref. 8). Other changes are made to reflect revisions to the core design parameters such as shutdown margin, boron and control rod worth, axial and radial peaking factors, and cycle length, as well as numerous editorial changes.

During the review, the staff also identified a few discrepancies associated with administrative changes. In response to the staff's request for additional information (Ref. 2), the licensee provided further changes to Revision 1 of the Topical report. These modifications include clarifications to revised sections and minor changes to equations. The NRC staff has reviewed the analyses associated with the changes to Topical Report DPC-NF-2010A and the responses to the requests for additional information pertaining to these changes. The staff has concluded

that the changes to this topical report consist mostly of administrative changes and clarifications to the original NRC approved topical report and that there are no unreviewed methodology or regulatory issues. Therefore, the staff finds the changes acceptable.

3.1.3 DPC-NE-2011, Revision 1

Topical Report DPC-NE-2011, (Ref. 9), describes the methodology for performing a maneuvering analysis for four-loop plants, such as McGuire and Catawba Nuclear Station. The licensee has developed this methodology as an alternate to the existing Relaxed Axial Offset Control Methodology. The licensee pointed out that this maneuvering analysis results in several advantages: more flexible and prompt engineering support for the operating stations, consistency with the methods of the licensee's nuclear design process, and potential increases in available margin through the use of three-dimensional monitoring techniques. The increase in margin occurs in limits on power distribution, control rod insertion, and power distribution inputs to the overpower delta-temperature and over-temperature delta-temperature reactor protection system trip functions.

Revision 1 of DPC-NE-2011, updates the report to include editorial changes, and to permit the use of certain methods approved subsequent to the implementation of the original version, such as the use of CASMO-3/SIMULATE-3P methodology (Ref. 8). Other changes are made to reflect revisions to the core design parameters such as power peaking factors, axial and radial power distributions, and cycle length, as well as numerous editorial changes.

In response to the NRC staff's request for additional information (Ref. 2), the licensee provided additional information to the staff regarding cycle depletion times to clarify issues associated with power peaking versus burnup as a function of cycle time. The licensee's amendment request also included clarifications to revised sections and minor changes to equations. The NRC staff has reviewed the analyses associated with the changes to Topical Report DPC-NE-2011-A and the responses to the requests for additional information pertaining to the requested changes. Since the changes to this topical report consists mostly of administrative changes and clarifications to the onginal NRC approved topical report, the staff find the changes acceptable.

3.1.4 DPC-NE-1003, Revision 1

Topical Report DPC-NE-1003 (Ref. 10) describes the measurement procedure used to determine the inferred bank worth and the calculation procedures used to develop the rod swap correction factor that accounts for the effect of a test bank on the partial integral worth of the reference bank. The NRC approved the report in May 1987 (Ref. 11) for rod worth measurement of reload cores for McGuire and Catawba Stations, Units 1 and 2.

Revision 1 of DPC-NE-1003 updates the report to permit the use of certain methods approved subsequent to the implementation of the original version, such as the use of CASMO-3/ SIMULATE-3P reactor physics methods (Ref. 8). Other changes are made to reflect the revision of the rod swap measurement procedures, and various editorial changes. In response to staff questions, the licensee, in its letter of August 7, 2002, provided the current version of the control rod worth measurement rod swap procedures, PT/0/A/4150/11A, dated January 19, 1996. The staff review of this current control rod worth measurement procedure has found it acceptable. The licensee in the August 7, 2002, letter also modified the equation in Section 3

of the topical report for the calculation of the inferred rod bank worth from the measured reference bank worth and bank height. This change is consistent with the equation described in step 12.12.5 of the current measurement procedures of January 19, 1996. Therefore, Revision 1 of DPC-NE-1003, as modified in the August 7, 2002, letter, is acceptable.

3.2 Proposed TS Changes

This section addresses the staff's evaluation of the proposed changes to TS 5.6.5.a regarding the cycle-specific operating parameters specified in the COLR. The staff review of these TS changes are based on the guidance of GL 88-16.

TS 5.6.5.a provides a list of core operating limits that are established prior to each reload cycle, or prior to any remaining portion of a reload cycle. The valves of the limits are in the COLR. For Catawba Units 1 and 2, the licensee proposed to revise the list by:

(1) adding "60 ppm" to Item 5.6.5.a.1 regarding the moderator temperature coefficient (MTC) surveillance limit for Specification 3.1.3,

(2) adding Item 5.6.5.a.12, "31 EFPD surveillance penalty factors for Specifications 3.2.1 and 3.2.2," and

(3) adding Item 5.6.5.a.13, "Reactor makeup water pumps combined flow rates limit for Specifications 3.3.9 and 3.9.2."

These changes are evaluated below.

3.2.1 MTC 60 ppm Surveillance Limit

Catawba TS LCO 3.1.3 specifies that the MTC be maintained within the LCO limits, which are based on the safety analysis assumptions. For verification that these LCO limits are met, the Surveillance Requirements of TS 3.1.3 also places surveillance limits for conducting the end of cycle MTC measurement at 300 ppm and 60 ppm boron concentration. The LCO limits and the 300-ppm and 60-ppm surveillance limits are specified in the COLR. However, TS Item 5.6.5.a.1 operating limits does not currently identify the 60-ppm surveillance limit.

The proposed change to the Catawba TS would add the 60-ppm surveillance limit in Item 5.6.5.a.1. The new TS would read "Moderator Temperature Coefficients BOL and EOL limits and 60 ppm and 300 ppm surveillance limit for Specification 3.1.3." The NRC approved incorporating the 60-ppm surveillance limits into the COLR during the Improved Technical Specifications conversion in 1998 (Ref. 12 and 13); however, reference to this surveillance was not included in TS Item 5.6.5.a.1 at that time. The proposed TS change to include the 60-ppm surveillance limit in TS Item 5.6.5.a.1 provides consistency with previously approved requirements and, therefore, it is acceptable.

3.2.2 Relocation of Hot Channel Factors Surveillance Penalty Factors to COLR

Surveillance Requirements in TS 3.2.1 and 3.2.2, respectively, require that the heat flux hot channel factor, F_q (x,y,z), and the enthalpy rise hot channel factor, $F_{\Delta h}$ (x,y), be measured every 31 effective full power days (EFPD) during equilibrium conditions using the incore detector

system to verify they are within the respective limits. To address the possibility that these hot channel factors may increase and exceed their allowable limits between surveillances, penalty factors are applied to these hot channel factors if their margins to the respective limits have decreased since the previous surveillance. These margin-decrease penalty factors are calculated by projecting the limiting hot channel factors over the 31 EFPD surveillance intervals with the maximum changes at the limiting core location, and are based on reload core design. In Section 8, "Improved Technical Specification Changes," of DPC-NE-2009, the licensee proposed to replace the penalty factors with tables of penalty value as functions of burnup in the COLR to facilitate cycle-specific updates. TS Item 5.6.5.b.14 lists topical report DPC-NE-2009-P-A that includes (in response to a staff question during the review of DPC-NE-2009) the approved methodology used to calculate these burnup-dependent penalty factors. The staff found the methodology and the inclusion of the burnup-dependent margin decrease penalty factors in the COLR acceptable as stated in the staff's safety evaluation supporting license amendment Nos. 180 and 172, respectively for Catawba Units 1 and 2 (Ref. 15).

The proposed changes to the Catawba TS would add Item 5.6.5.a.12, that reads: "31 EFPD surveillance penalty factors for Specifications 3.2.1 and 3.2.2." The addition of TS Item 5.6.5.a.12 would make it consistent with the previous staff approval of including these surveillance penalty factors in the COLR and, therefore, this proposed change is acceptable.

3.2.3 Reactor Makeup Water Pumps Combined Flow Rates Limit

The relocation of the reactor makeup water pumps combined flow rates limit for the boron dilution mitigation system from Catawba TS 3.3.9 and 3.9.2 to the COLR was approved by the NRC as described in a letter dated March 25, 1994 (Ref. 16). The reactor makeup water pumps flow rate limit is included in the Catawba COLR.

The proposed changes to the Catawba TS would add Item 5.6.5.a.13, "Reactor makeup water pumps combined flow rates limit for Specification 3.3.9 and 3.9.2," to TS 5.6.5.a. The addition of this item would make the TS 5.6.5.a list consistent with the core operating limits included in the Catawba COLR and is therefore, acceptable.

4.0 SUMMARY

The staff has reviewed the revisions of four previously approved topical reports described in Section 1.0 of this Safety Evaluation, and the proposed changes to Catawba Nuclear Station, Units 1 and 2, TS 5.6.5.a related to the COLR. Based on our evaluation described in Section 3 of this Safety Evaluation, the staff concludes that the these topical report revisions, as amended by the August 7, 2002, letter, and the TS changes are acceptable.

5.0 STATE CONSULTATION

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

6.0 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 Commission has previously issued a proposed finding that the amendments involve no significant hazards consideration, and there has been no public comment on such finding [67 FR 54680]. 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.

7.0 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.

8.0 REFERENCES

1. Letter from M. S. Tuckman, Duke Energy Corporation, to US Nuclear Regulatory Commission, "Duke Energy Corporation; Catawba Nuclear Station Units 1 and 2, Docket Nos. 50-413, 50-414; McGuire Nuclear Station Units 1 and 2, Docket Nos. 50-369, 50-370; License Amendment Request Applicable to Technical Specifications 5.6.5, Core Operating Limits Report; Revisions to BASES 3.2.1 and 3.2.3; and Revisions to Topical Reports DPC-NE-2009-P, DPC-NF-2010, DPC-NE-2011-P, and DPC-NE-1003," October 7, 2001.

2. Letter from M. S. Tuckman, Duke Energy Corporation, to US Nuclear Regulatory Commission, "Duke Energy Corporation; McGuire Nuclear Station Units 1 and 2, Docket Nos. 50-369 and 370; Catawba Nuclear Station Units 1 and 2, Docket Nos. 50-413 and 414; Response to NRC Request for Additional Information - TAC nos. MB3222, MB3223, MB3343 and MB3344) and License Amendment Request Supplement," August 7, 2002.

3. Letter from M. S. Tuckman, Duke Energy Corporation, to US Nuclear Regulatory Commission, "License Amendment Request Applicable to the Technical Specifications Requirements for the Core Operating Limits Report - Oconee, McGuire, and Catawba Technical Specifications 5.6.5," December 20, 2001.

4. Letter from Dennis Crutchfield, USNRC, to All Power Reactor Licensees and Applicants, "Removal of Cycle-Specific Parameter Limits from Technical Specifications (Generic Letter 88-16)," October 4, 1988.

5. DPC-NE-2009-P-A, "Duke Power Company Westinghouse Fuel Transition Report," December 1999.

6. Letter from J. S. Galembush, Westinghouse Electric Company, to US Nuclear Regulatory Commission, "1998 Annual Notification of Changes to the Westinghouse Small Break LOCA and Large Break LOCA ECCS Evaluation Models, Pursuant to 10 CFR 50.46(a)(3)(ii)," NSD-NRC-99-5839, July 15, 1999.

7. DPC-NF-2010A, "Duke Power Company McGuire Nuclear Station and Catawba Nuclear Station Nuclear Physics Methodology for Reload Design," June 1985.

8. DPC-NE-1004A, Revision 1, "Nuclear Design Methodology Using CASMO-3/ SIMULATE-3P," SER dated April 26, 1997.

9. DPC-NE-2011, "Nuclear Design Methodology Report for Core Operating Limits of Westinghouse Reactors," March 1990.

10. DPC-NE-1003, "Rod Swap Methodology Report for Startup Physics Testing," December 1986.

11. Letter from Darl Hood, USNRC, to H. B. Tucker, Duke Power Company, "Rod Swap Methodology Report for Startup Physics Testing, McGuire and Catawba Nuclear Stations, Units 1 and 2 (TACs 62981, 62982, 62983, 62984)," May 22, 1987.

12. Letter from Frank Rinaldi, USNRC, to H. B. Brown, McGuire Site, Duke Energy Corporation, "Issuance of Amendments - McGuire Nuclear Station, Units 1 and 2, (TAC Nos. M98964 and M98965)," September 30, 1998.

13. Letter from Peter Tam, USNRC, to G. R. Peterson, Catawba Nuclear Station, Duke Energy Corporation, "Issuance of Amendments - Catawba Nuclear Station, Units 1 and 2 (TAC Nos. M95298 and M95299)," September 30, 1998.

14. Letter from Frank Rinaldi, USNRC, to H. B. Brown, McGuire Site, Duke Energy Corporation, "McGuire Nuclear Station, Units 1 and 2, Re: Issuance of Amendments (TAC Nos. MA2411 and MA2412)," September 22, 1999.

15. Letter from Peter Tam, USNRC, to G. R. Peterson, Catawba Nuclear Station, Duke Energy Corporation, "Catawba Nuclear Station, Units 1 and 2, Re: Issuance of Amendments (TAC Nos. MA2359 and MA2361)," September 22, 1999.

16. Letter from Robert F. Martin, USNRC, to David L. Rehn, Catawba Site, Duke Power Company, "Issuance of Amendments - Catawba Nuclear Station, Units 1 and 2 Cycle Specific Parameters to the Core Operating Limits Report (TAC Nos. M85472 and M85473)," March 25, 1994.

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