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U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

Attention: Document Control Desk

Subject: McGuire Nuclear Station
Docket Numbers 50-369 and -370
Catawba Nuclear Station
Docket Numbers 50-413 and -414
Duke Power Company Topical Report DPC-NE-1004;
Supplemental Information to Support Minor Revision

By letter dated December 12, 1995, Duke Power Company proposed a minor revision to Topical Report DPC-NE-1004, "Nuclear Design Methodology Using CASMO-3/SIMULATE-3P." The NRC staff requested that additional information be provided, by letter dated February 28, 1996. Attached are responses to questions in the staff's RAI.

If there are additional questions, or more information is needed, please call Scott Gewehr at (704) 382-7581.

Very truly yours,

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Response to NRC Questions

1. What is the reason for the increase from 12 to 24 axial nodes? DPC's letter of December 12, 1995 mentions several factors potentially involved with the change, but the reason for the change is not explicitly discussed.

   Answer: The choice of a 12 axial node SIMULATE-3 model was originally chosen as a compromise between the desire for more axial nodes and that of computer run time. Since initially submitting DPC-NE-1004, the increase in computational efficiency has allowed Duke to increase the number of nodes without a substantial increase in computer cost. Additional reasons for wanting to increase the number of axial nodes in SIMULATE-3 from 12 to 24 are discussed below.

   a. SIMULATE-3 has a coding limitation on the number of axial regions which can be modelled within one node. For axial blanket fuel with burnable poisons rods, the nodal length must be decreased to less than 12 inches in order to satisfy this criteria. Axial node segments of 6 or 8 inches satisfy this requirement. A 6 inch axial node segment was chosen because this causes the nodal boundaries to match up exactly with the transition from the blanketed fuel region to the non-blanketed fuel region. Modelling these boundaries exactly increases calculational accuracy in this region of the core since cross sections of two different enrichments are not averaged.

   b. To better account for the axial dependence in predicted and measured power distributions of current generation vendor fuel designs which are not axially homogeneous (eg. axial blanket fuel).

2. Discuss all of the input involved in the changes shown in the Observed Nuclear Reliability Factors (ONRF). Are the changes the result of an analysis of actual fuel cycle data versus previously assumed values? If so, which is more conservative? Or, are the changes the result of the axial noding change.

   Answer: The ONRF's for both 12 and 24 axial level SIMULATE-3 models are the result of analysis of actual fuel cycles. The changes in the calculated ONRF's are the result of using power distributions from more recent fuel cycles, increasing the number of axial nodes from 12 to 24 and accounting for the axial dependency of the power to reaction rate ratio in the measured power distribution. The original ONRF's for Westinghouse plants were developed in DPC-NE-1004 based on the analysis of the McGuire 2 Cycle 4, McGuire 2 Cycle 5, Catawba 1 Cycle 3 and Catawba 2 Cycle 2 core designs. The 24 axial level ONRF's were developed based on current generation core designs in order to reflect the more aggressive reload design strategies reflected in current core designs. Specifically, the power distribution database used to develop the 24 axial level ONRF's included the effects of higher enriched fuel, longer cycle lengths, higher burnup, higher BPRA loadings, and axial blankets. Characteristics of the fuel cycles analyzed are shown in Table 1.

Table 1
Fuel Cycle Characteristics

Cycle	No. of Feed and Enrichment	Design EFPD	Cycle Characteristics
M1C09	64 feed at 3.45 w/o	340	Low number of feed and BP loading
M2C09	76 feed at 3.65 w/o	395	High Enr. and BP loading
C1C07	72 feed at 3.45 w/o	350	Typical
C2C06	76 feed at 3.75 w/o	380	First Transition cycle to Mk-BW
C2C07	40 feed at 4.0 w/o + 8 feed at 3.60 w/o+ 40 feed at 3.50 w/o	430	Axial Blankets and High No. of feed

+ Axial Blanket fuel with 6 inch natural uranium blankets

The 24 axial level Fq and Fz ONRF's decreased by 2.0 and 2.2%, respectively, relative to DPC-NE-1004 values. However, the 24 axial level model ONRF for FΔH increased by 0.3% over the DPC-NE-1004 value. This increase is considered statistically insignificant and is attributed to the selection of more challenging, and radially heterogeneous core designs for the creation of the 24 axial level statistical data base relative to the core designs which were available when the DPC-NE-1004 FΔH ONRF was developed. The significance of the 0.3% increase in the FΔH ONRF is addressed in the answer to question 3.

Answer to Question 2 (Continued)

The decrease in the Fq and Fz uncertainties is attributed to the increase in axial resolution of the core model resulting from the axial node size reduction and from the use of axially dependent power to reaction rate ratios (which better characterize the spectral dependency and axial geometry of the fuel) to process measured reaction rates. These factors result in the reduction in the bias term included in the ONRF derivation. Note also that the variability of the statistical population as measured by the standard deviation of the Fq and Fz populations remain similar to previously calculated values. Therefore, the reduction in the Fq and Fz uncertainty factors is almost entirely due to the reduction in the predicted to measured bias.

The statistical data used to develop the 12 and 24 axial level ONRF's is provided in Table 2. The equations used to develop the ONRF's are contained in Section 4.2 of DPC-NE-1004.

Table 2
12 and 24 Axial Level
Observed Nuclear Reliability Factors

12 Axial Level ONRF's:

Parameter	N	$\overline{M}$	$\overline{D}$	k	S(D)	ONRF
FΔH	1455	1.145	0.000	1.713	0.011	1.017
Fq	1998	1.257	-0.027	1.703	0.026	1.057
Fz	2520	1.148	-0.027	1.697	0.020	1.053

24 Axial Level ONRF's:

Parameter	N	$\overline{M}$	$\overline{D}$	k	S(D)	ONRF
FΔH	2516	1.168	0.0034	1.6973	0.016	1.020
Fq	3162	1.281	-0.0055	1.6911	0.025	1.037
Fz	4200	1.138	-0.0103	1.6854	0.015	1.031

3. Please identify which transients and accidents are affected by each change. Provide supporting analysis results for the limiting transients and accidents demonstrating how their acceptance criteria (DNB, kw/ft, pressure, etc.) are met.

Answer: Future safety analyses will use FΔH, Fq and Fz uncertainties with values greater than or equal to the values of the uncertainty factors shown in Table 3. The uncertainties calculated in Table 3 were developed using the same statistical data used in the development of ONRF's and are based on the FΔH, Fq, and Fz uncertainty factor equations developed in Section 5.1 through 5.3 of DPC-NE-1004. The difference between the ONRF's shown in Table 2 and the uncertainties shown in Table 3 is the statistical combination of the pin uncertainty with the assembly uncertainty.

Table 3
DPC-NE-1004 and 24 Axial Level Uncertainties
Without Engineering Hot Channel Factor

Parameter	Bias	Assembly Uncertainty	Pin Uncertainty	Total Uncertainties	DPC-NE-1004 Uncertainties +
FΔH	-0.0029	0.0234	0.02	1.028	1.026
Fq	0.0043	0.0329	0.02	1.043	1.061
Fz	0.0091	0.0216	----	1.031	1.053

+ 12 axial level uncertainty

The Fq and Fz uncertainty factors used in current and previous accident analyses bound values calculated for the 24 axial level model. Therefore, there is no impact to past, present or future safety analyses in which only Fq and Fz uncertainties are used. The increase in the FΔH uncertainty over the topical value is of no safety concern for future analyses because a value greater than or equal to the 24 axial level FΔH uncertainty will be used in these safety analyses.

Since the increase in the FΔH uncertainty factor may be a result of the analysis of more complex reactor cores and not a result of transitioning to a 24 axial level model, the impact of this increase was assessed by performing a review of FSAR Chapter 15 accidents and their appropriate acceptance criteria which could be affected by an increase in radial (FΔH) uncertainty factor. The following calculations are affected:

- Pin Pressure
- Creep collapse
- DNB

The calculation of peak fuel enthalpy, Linear Heat Rate to Melt (LHRTM) kw/ft limits and primary and secondary peak pressures are not affected by the increase in radial uncertainty for the following reasons.

Answer to Question 3 Continued

Significant margin exists to the peak fuel enthalpy limit of 280 cal/gm, such that this parameter is not limiting. For the calculation of LHRTM kw/ft limits, a total (Fq) uncertainty and not a radial uncertainty is applied since this is local phenomenon. Primary and secondary system peak pressure response calculations are based on a balance between energy removed by the steam generators, and energy added from the reactor core. Since the pressure response is dependent on the rate of energy deposited from the reactor core, independent of the peaking within the core, local peaking uncertainties are not important. Therefore, the calculation of accident acceptance criteria and confirmation of limits for peak fuel enthalpy, LHRTM and primary and secondary side peak pressure are unaffected.

Peak pin pressure and creep collapse calculations assume a bounding radial uncertainty of 1.036. Since this uncertainty bounds the 24 axial level value of 1.028, it can be concluded that the current and past analyses are unaffected.

For the FSAR Chapter 15 accidents in which DNB is a concern, thermal analyses are performed to ensure that fuel clad integrity is maintained by ensuring that the minimum DNBR remains above the 95/95 DNBR limit. Two types of DNBR analyses are performed. Thermal analyses which are based on the Statistical Core Design (SCD) methodology described in reference 1, and thermal analyses which are not based on this methodology (non-SCD DNBR analyses). The FSAR Chapter 15 accidents which are based on the SCD methodology are unaffected by the change in uncertainty factors. This is because radial and axial uncertainty factors of 1.04 and 1.053 are assumed in these accident analyses, which bound the 24 axial level uncertainty factors.

The FSAR Chapter 15 accidents which are based on the non-SCD methodology, along with the radial and axial uncertainty factors assumed in the analysis of each accident, are summarized in Table 4.

Table 4
Non-SCD DNB Uncertainty Factors

Accident	Radial Uncertainty	Axial Uncertainty
Startup of an Inactive RC Pump at an Incorrect Temp.	N/A	N/A
Steam Line Break	1.036	1.053
Locked Rotor	1.026	1.053
Rod Ejection	1.026	1.053

From the data in Table 4 it is not immediately evident that past DNBR analyses performed based on non-SCD methodology were conservative. Therefore, each of these analyses are discussed below.

Answer to Question 3 Continued

The startup of an inactive coolant pump at an incorrect temperature transient is a non-limiting transient which is bounded by the analysis of other FSAR transients. The FΔH uncertainty assumed in the steam line break accident bounds both the 12 and 24 axial level uncertainties. For the Locked Rotor and Rod Ejection accidents, the FΔH uncertainty assumed in the accident analyses does not bound the FΔH uncertainty calculated for the 24 axial level model. However, since DNB is a function of both the radial and axial power distribution, the decrease in the 24 level axial uncertainty factor more than offsets the slight increase in radial uncertainty, resulting in an increase in DNB margin. Therefore, it can be concluded that previous accident analyses performed are conservative and the consequences of FSAR accidents previously evaluated and the margin to safety as defined in Technical Specifications is not reduced for the Locked Rotor and Rod Ejection accidents. In addition, it should be noted that margin retained between the 95/95 correlation and design DNBR limit used in accident analyses, (which is retained to account for unanticipated non-conservatisms) could also have been used to account for the slight increase in radial uncertainty.

In summary, the 24 axial level calculational uncertainty factors will be used in all safety related analyses in which a 24 axial level SIMULATE-3 model will be used. The slight increase in the FΔH uncertainty factor relative to the topical value does not increase the consequences or reduce the margin to safety of accidents previously evaluated. This is because either the accident was non-limiting, the accident analysis employed conservative FΔH uncertainty factors, or the tradeoff between the decrease in Fz uncertainty more than compensated for the increase in FΔH uncertainty. The calculation of accident acceptance criteria for LHRTM, peak fuel enthalpy and peak pressure are also unaffected by the increase in radial uncertainty factor. Therefore, it can be concluded that the consequences of FSAR accidents previously evaluated and margin to safety as defined in the bases to Technical Specification is not decreased.

References

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