Duke Power Company
P.O. Box 1006
Charlotte, NC 28201-1006

M. S. Tuckman
Senior Vice President
Nuclear Generation
(704)382-2200 Office
(704)382-4360 Fax

DUKE POWER

March 15, 1996

U. S. Nuclear Regulatory Commission
Attention: Document Control Desk
Washington, DC 20555

Subject: Catawba Nuclear Station, Units 1 and 2,
Docket Nos. 50-413 and 414
McGuire Nuclear Station, Units 1 and 2,
Docket Nos. 50-369 and 370
Response to Request for Additional Information

By letters dated September 30, 1994 and September 18, 1995, Duke Power Company submitted proposed revisions to the Catawba Unit 1 and McGuire Units 1 and 2 Technical Specifications. The proposed changes were related to replacement of the steam generators for each unit.

By letter dated February 21, 1996, the NRC Staff provided a Request for Additional Information (RAI) related to the proposed Technical Specification changes for Catawba Unit 1. With the similarities in the McGuire and Catawba submittals and our desire to have the Technical Specifications approved for all three units by June 12, 1996, responses to the RAI are provided for all three units in Attachment 1.

Since the September 1994 submittals, other Technical Specification changes have been issued affecting many of the same pages. For clarity, the Technical Specification changes requested in the September 1994 and September 1995 submittals have been combined and resubmitted in Attachments 2 through 7. Marked-up and typed copies of the combined Catawba Units 1 and 2 Technical Specification pages are provided in Attachments 2 and 3. Mark-up and typed copies of the McGuire Units 1 and 2 Technical Specification are provided in a separate unit format in Attachments 4 through 7.

My letter of January 19, 1996 submitted proposed changes to Catawba Technical Specification 6.9.1.9 that would reflect the NRC's approved revisions to Topical Reports DPC-NE-3000 and DPC-NE-3002. Similar changes were requested in the September 30, 1994 submittals and have been updated and included in this submittal for Catawba Units 1 and 2 and

A001
1/1

McGuire Units 1 and 2.

As previously requested in the September 1994 and September 1995 submittals, Duke Power is requesting that the Catawba Unit 1 and the McGuire Units 1 and 2 changes be reviewed and approved concurrently with the Catawba Unit 1 steam generator replacement outage and that the changes become effective upon replacement of the steam generators. The Catawba Unit 1 outage is currently scheduled to begin on June 13, 1996.

If any additional information is required, please call Robert Sharpe at (704) 382-0956.

Very truly yours,

M. S. Tuckman

M. S. Tuckman

Attachments

xc:

S. D. Ebneter
Regional Administrator, Region II
U. S. Nuclear Regulatory Commission
101 Marietta Street, NW, Suite 2900
Atlanta, GA 30323

R. J. Frudenburger
Senior Resident Inspector
Catawba Nuclear Station

G. F. Maxwell
Senior Resident Inspector
McGuire Nuclear Station

R. E. Martin
Project Manager, ONRR

V. Nerses
Project Manager, ONRR

ATTACHMENT 1
RESPONSES TO RAI

Question 1

With respect to the proposed changes in steam generator (SG) low-low and high-high water level setpoints and the change in nominal average reactor coolant temperature, and the change in reactor coolant system volume, please identify which transients and accidents are affected from each TS change. Identify the most limiting transient or accident for the specified TS change. Provide supporting analysis results for the limiting case demonstrating how it meets the acceptance criteria with respect to system pressure and fuel performance (DNB, etc.).

Response

The following thermal-hydraulic system transients are reanalyzed in order to ensure that the acceptance criteria continue to be met for any cases in which the feedring steam generator might result in a more severe challenge to the criteria. None of the accidents are reanalyzed explicitly due to the impact of the individual Technical Specifications changes. Both the accident reanalyses and the Technical Specifications changes are resultant from the design differences between the existing preheat steam generators and the feedring steam generator.

All analyses are performed consistent with either NRC-approved methodology or methodology submitted for approval. Listed below are the applicable Duke Power Company analysis methodology topical reports:

- DPC-NE-3000-P, Thermal-Hydraulic Transient Analysis Methodology, Duke Power Company, Oconee, McGuire and Catawba Nuclear Stations, Revision 1, SER dated December 1995.

- DPC-NE-3001-PA, Multidimensional Reactor Transients and Safety Analysis Physics Parameters Methodology, Duke Power Company, McGuire and Catawba Nuclear Stations, November 1991.

- DPC-NE-3002, FSAR Chapter 15 System Transient Analysis Methodology, Duke Power Company, McGuire and Catawba Nuclear Stations, Revision 1, SER dated December 1995.

- DPC-NE-3004-P, Mass and Energy Release and Containment Response Methodology, Duke Power Company, McGuire and Catawba Nuclear Stations, SER dated September 1995.

In addition, the steam generator tube rupture and LOCA reanalyses rely upon the following vendor topical reports:

- WCAP-10698-PA, "SGTR Analysis Methodology to Determine the Margin to Steam Generator Overfill," Westinghouse Electric Corporation.
- BAW-10168, "BWNT Loss-of-Coolant Accident Evaluation Model for Recirculating Steam Generator Plants," Revision 3, Babcock & Wilcox.
- BAW-10174, "Mark-BW Reload LOCA Analysis for the McGuire and Catawba Units," Revision 1, Babcock & Wilcox.

A. Mass and energy release for postulated loss of coolant accidents inside containment (6.2.1.3)

This event is analyzed to ensure that the peak containment pressure limit is not exceeded. The analysis is performed in accordance with the analytical model and methodology described in topical report DPC-NE-3004. The respective peak lower containment pressures calculated for Catawba Unit 1, Catawba Unit 2 and McGuire (both units) are 11.77, 10.29 and 12.48 psig, all of which meet the 15 psig acceptance criterion.

B. Mass and energy release for postulated secondary system pipe ruptures inside containment (6.2.1.4)

This event is analyzed to ensure that the peak containment temperature limit is not exceeded. The analysis is performed in accordance with the analytical model and methodology described in topical report DPC-NE-3004. The peak temperature in the break compartment is calculated to be 313°F, which does not exceed the environmental qualification (EQ) limit of 340°F.

C. Feedwater system malfunction causing an increase in feedwater flow (15.1.2)

This ANS Condition II event is analyzed to show that DNB does not occur. The analysis is performed in accordance with the analytical model and methodology described in topical reports DPC-NE-3000 and DPC-NE-3002. The minimum DNBR is calculated as 1.86, which is well above the 1.55 design limit.

D. Excessive increase in secondary steam flow (15.1.3)

This ANS Condition II event is analyzed to show that DNB does not occur. The analysis is performed in accordance with the analytical model and methodology described in topical reports DPC-NE-3000 and DPC-NE-3002. This transient analysis showed that the reactor reached an equilibrium condition that did not challenge the OPΔT or OTΔT reactor trip functions, which are designed to protect the core against DNB. Therefore, an explicit DNBR calculation was not performed.

E. Inadvertent opening of a steam generator relief or safety valve (15.1.4)

This ANS Condition II event is reanalyzed to confirm that the transient response is bounded by the steam system piping failure event. The analysis is performed in accordance with the analytical model and methodology described in topical reports DPC-NE-3000, DPC-NE-3001, and DPC-NE-3002. The peak thermal power level reached was less than half of that resulting from the steam system piping failure event. Therefore, an explicit DNBR calculation was not performed.

F. Steam system piping failure (15.1.5)

This ANS Condition III & IV event is analyzed to the more stringent Condition II criterion of ensuring that DNB does not occur. The analysis is performed in accordance with the analytical model and methodology described in topical reports DPC-NE-3000 and DPC-NE-3001. The minimum DNBR is calculated to be 1.78, which is well above the 1.45 design limit.

G. Turbine trip (15.2.3)

This ANS Condition II event is analyzed to show that peak primary and secondary system pressures do not exceed the applicable limits. The analysis is performed in accordance with the analytical model and methodology described in topical reports DPC-NE-3000 and DPC-NE-3002. The calculated peak primary and secondary pressures are 2674.3 and 1285.7 psig, respectively. The corresponding acceptance criteria are 2733.5 psig (110% of 2485) and 1303.5 psig (110% of 1185).

H. Loss of non-emergency AC power to the station auxiliaries (15.2.6)

This ANS Condition II event is reanalyzed to demonstrate the adequacy of the natural circulation cooling in the modified reactor coolant loop configuration. The analysis is performed in accordance with the analytical model and methodology described in topical reports DPC-NE-3000 and DPC-NE-3002. The results of this analysis show that, within 10 minutes of the loss of offsite power, a stable natural circulation flow rate of approximately 5% of the full power value is established with a core ΔT of less than 30°F.

I. Loss of normal feedwater flow (15.2.7)

This ANS Condition II event is reanalyzed to confirm that the transient response is bounded by the turbine trip event. The analysis is performed in accordance with the analytical model and methodology described in topical reports DPC-NE-3000 and DPC-NE-3002. The position presented in DPC-NE-3002, that this transient is bounded by turbine trip, remains valid.

J. Feedwater system pipe break (15.2.8)

This ANS Condition IV event is analyzed to demonstrate the capability of the degraded secondary system to effectively cool the reactor core. The analysis is performed in accordance with the analytical model and methodology described in topical reports DPC-NE-3000 and DPC-NE-3002.

The adequacy of the long-term core cooling capability is verified by the prevention of hot leg boiling. A minimum subcooling of 17.5°F was reached during the overheating phase of the feedline break transient.

K. Reactor coolant pump shaft seizure - locked rotor (15.3.3)

This ANS Condition IV event is analyzed to calculate the percentage of fuel rods that experience DNB and to subsequently determine the offsite radiological consequences. The analysis is performed in accordance with the analytical model and methodology described in topical reports DPC-NE-3000 and DPC-NE-3002.

The offsite dose results (at the EAB) are summarized in the table below. None of the calculated doses exceed a small fraction (10%) of the 10CFR100 limits.

	McGuire 1 & 2	Catawba 1	10CFR100 Limit
Whole body dose	0.624	0.252	2.5
Thyroid dose (pre-existing iodine spike)	22.65	8.294	30

L. Steam generator tube rupture (15.6.3)

This ANS Condition IV event is analyzed to show that a) DNB does not occur, b) the calculated offsite doses do not exceed the acceptance criterion, and c) SG overfill is avoided. The analysis is performed in accordance with the analytical model and methodology described in topical reports DPC-NE-3000, DPC-NE-3002, and WCAP-10698-PA.

The minimum DNBR was calculated to be 1.93, which is well above the 1.55 design limit. The offsite dose results (at the EAB) are summarized in the table below. None of the calculated doses exceed the applicable 10CFR100 limits. Lastly, the time to overfill for the replacement generators was determined to be bounded by that for the Model D steam generators, which has been previously proven acceptable.

	McGuire 1 & 2	Catawba 1	10CFR100 Limit
Whole body dose	0.342	0.126	2.5
Thyroid dose:			
pre-existing iodine spike	38.2	53.4	300
coincident iodine spike	14.9	20.1	30

M. Loss of coolant accidents (15.6.5)

The methodology employed in the LOCA analysis is in accordance with 10CFR50.46 and 10CFR50 Appendix K and is documented in topical reports BAW-10174 and BAW-10168P.

The maximum cladding temperature calculated for a large break is 2075°F, which is less than the Acceptance Criteria limit of 2200°F of 10CFR 50.46. The maximum local metal water reaction is 5.9 percent, which is well below the embrittlement limit of 17 percent as required by 10CFR 50.46. The total core metal water reaction is calculated to be 0.61 percent as compared with the 1 percent criterion of 10CFR 50.46, and the cladding temperature transient is terminated at a time when the core geometry is still amenable to cooling. As a result, the core temperature will continue to drop and the ability to remove decay heat generated in the fuel for an extended period of time will be provided.

The maximum calculated peak cladding temperature for the small breaks analyzed is 1200°F. This result is well below all Acceptance Criteria limits of 10CFR 50.46 and no case is limiting when compared to the results presented for large breaks.

N. Postulated secondary system pipe rupture outside containment

This event is analyzed to ensure that the Doghouse equipment qualification temperature limit is not exceeded. The analysis is performed in accordance with the analytical model and methodology described in topical report DPC-NE-3004. A comparison of the mass and energy release rates showed that the reanalyzed transient results were bounded by the existing vendor-supplied analysis. Therefore, an explicit doghouse temperature response calculation was not performed.

Transients Reanalyzed For Offsite Dose Only

A. Single RCCA withdrawal (15.4.3.d)

The offsite dose results (at the EAB) are summarized in the table below. None of the calculated doses exceed a small fraction (10%) of the 10CFR100 limits.

	McGuire 1 & 2	Catawba 1	10CFR100 Limit
Whole body dose	0.228	See Note	2.5
Thyroid dose (pre-existing iodine spike)	26.97	"	30

Note: The input into the offsite dose calculation for the single uncontrolled RCCA withdrawal is conservatively bounded by those for the locked rotor w/o LOOP. The locked rotor doses are significantly below the applicable 10CFR100 limit. Therefore, no specific dose analysis was performed for the single RCCA withdrawal event.

B. Spectrum of RCCA ejection accidents (15.4.8)

The offsite dose results (at the EAB) are summarized in the table below. All doses are well within (25%) the 10CFR100 limits.

	McGuire 1 & 2	Catawba 1	10CFR100 Limit
Whole body dose	0.939	1.127	6
Thyroid dose (no iodine spike)	43.79	60.23	75

Question 2

SR 4.4.5.4.b, TS page 3/4 4-16a

The standard TS reads as follows: "The steam generator shall be determined OPERABLE after completing the corresponding actions (plug all tubes exceeding the plug limit and all tubes containing through-wall cracks) required by Table 4.4-2." Duke Power did not include the phrase in the parentheses in its submittal. The staff believes that the phrase in the parentheses should be restored to the Catawba TS because the

replacement steam generators are not exempt from this requirement.

Response

See the revised mark-ups and typed copies of the Catawba TS provided as Attachments 2 and 3.

Question 3

BASES 3/4.4.5, TS page B 3/4 4-3a

Duke Power should delete the sentence that reads as follows: "Tubes experiencing outer diameter stress corrosion cracking within the thickness of the tube support plates are plugged or repaired by the criteria of 4.4.5.4.a.13." This statement is no longer applicable to Catawba Unit 1 after the replacement of the steam generators.

Response

See the revised mark-ups and typed copies of the Catawba TS provided as Attachments 2 and 3.

It was Duke Power's intent in revising the steam generator surveillance TS and BASES to remove references to "F*", "sleeve", or "repair" and related discussions, since these methodologies would not be applicable to the replacement steam generators. A few remaining references to these methodologies were identified and have been deleted as indicated in Attachments 2 through 7.