Duke Power Company
Catawba Nuclear Generation Department
4800 Concord Road
York, SC 29745

William R. McCollum, Jr.
Vice President
(803)831-3200 Office
(803)831-3426 Fax

DUKE POWER

April 10, 1997

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

Subject: Catawba Nuclear Station, Units 1 and 2
Dockets Nos. 50-413 and 50-414
Request for Additional Information Regarding the Operating License Amendment for the Steam Generator Tube Rupture Evaluation
(TAC Nos. M98107, M98108)

Per conference call on April 7, 1997, the NRC requested clarification of details regarding the additional information submitted by Duke Power to the NRC in a letter dated April 2, 1997 for the Steam Generator PORV Technical Specification Amendment. The proposed amendment revises Section 3/4.7.1.6 of the Technical Specifications to require four instead of three steam generator power operated relief valves (PORVs) and Section 15.6.3 of the Updated Final Safety Analysis Report (UFSAR) to require four instead of three PORVs and allow credit for local manual operation of the PORVs. The additional information requested is provided in the enclosure and should supply the clarification necessary to complete the amendment request.

We request that you review the additional information on a schedule consistent with the urgency of the original request. If you need additional clarification of the response to the questions or have additional questions please contact Martha Purser at (803)-831-4015.

Sincerely,

Mary L. Birch for

William R. McCollum, Jr.

A001/,

xc (with attachments):

L.A. Reyes, Regional Administrator, Region II

P.S. Tam, Senior Project Manager, ONRR

R.J. Freudenberger, Senior Resident Inspector, CNS

Response to Request for Additional Information for

Catawba Nuclear Station Units 1 and 2

Regarding Proposed Amendments for the Steam Generator PORVs

Reference: Phone conversation with the state NRR unit branch on April 7, 1997 regarding the additional information submitted by letter to the NRC dated April 2, 1997.

Question 1. List conservative assumptions used in the RETRAN analysis to justify 80 cu.ft. as an acceptable margin to overfill.

Response: Following is a list of conservative parameters used in the RETRAN analysis for overfill:

<u>Initial Power</u>
Per WCAP 10698, a full power condition without an overpower factor is conservative to maximize steam generator (SG) mass at the time of reactor trip. This was done in the Unit 2 analysis (Model D5 SGs). As discussed in our April 2, 1997 submittal, the BWI SGs have essentially a constant mass for power levels greater than ~25 %FP(full power). Since an overpower factor will not result in a decrease in SG mass at the time of reactor trip, the initial power level for the Unit 1 analysis (BWI SGs) conservatively applies an overpower factor (+2 %FP).

<u>Reactor Coolant System Pressure</u>
WCAP-10698 assumes a low initial (RCS) pressure to minimize the time to reactor trip on low RCS pressure. This allows for a turbine runback to occur while minimizing the time to reactor trip. The initial RCS pressure assumed in the Unit 1 (BWI SGs) analysis is adjusted high rather than low since turbine runback does not increase SG mass prior to reactor trip. This maximizes primary-to-secondary leakage. A high initial RCS pressure of 2295 psig is assumed, with nominal pressure being 2235 psig. It is assumed that operators manually trip the reactor in a conservatively short time after tube rupture initiation. This assumed time is less than the time it would have taken for reactor trip to occur on low pressurizer pressure with the initial RCS pressure adjusted low.

A low initial RCS pressure of 2193 psig (-42 psi) is assumed in the Unit 2 (Model D5 SGs) analysis. This minimizes the

time to reactor trip on low RCS pressure while allowing for a turbine runback to occur.

Pressurizer le

A high initi pressurizer level is used to maximize primary-to-secondary leakage. The full power program level (55%) is assumed with a 9% full span level uncertainty added to it.

Steam Generator Level

High secondary inventory at the time of reactor trip is conservative. As discussed in our April 2, 1997 submittal, a SG mass associated with 55% FP was assumed for the Unit 2 (Model D5) analysis. A narrow range level uncertainty of 13.6% full span (i.e., additional mass) was added to the program level (64.7%) for this power level.

For the Unit 1 analysis (BWI SGs), the program level for 100% FP is assumed (65%). A narrow range level uncertainty of 10.0% full span was added to this.

The difference in level uncertainty between the two SG types is due to velocity head effects around the pressure taps used by the level instrumentation.

Main Steam Safety Valves, Steam Line PORV Setpoints

Main Steam Safety valve (MSSV) and steam line PORV setpoints are adjusted low to maximize primary-to-secondary leakage. A -3% drift is used for the MSSVs while the steam line PORV setpoints are adjusted by -75 psi from nominal.

Auxiliary Feedwater

Conservatively high CA flowrates are used with the worst flow imbalance (i.e., the ruptured SG has the highest flow split). A conservatively fast CA start time is used (30 seconds) for both analyses. CA flow is throttled at the 0% FP narrow range level setpoint (39% and 62% for the BWI SGs and Model D5 SGs, respectively) plus uncertainty (4%).

Safety Injection

Conservatively high SI flowrates are used to maximize RCS pressure and tube leakage. A conservatively fast SI start time is used (12 seconds). For the Unit 1 (BWI SGs) analysis, manual safety injection is assumed at the time of reactor trip. For the Unit 2 analysis, safety injection occurs on low RCS pressure.

Break Location

The location of the break is on the cold side of the SG at the top of the tube sheet. This maximizes primary-to-secondary leakage.

Decay Heat

Conservatively high decay heat is modeled. Multipliers are applied to the code calculated decay heat such that for both cases 102% FP decay heat with 2 sigma uncertainty is predicted as a function of time after reactor trip.

Summary Tables For Overfill Analyses

Unit 1 (BWI SGs) Steam Generator Tube Rupture (SGTR) Overfill
Sequence of Events

Time (minutes)	Event Description
0.0	Double ended tube rupture initiation
3.0	Rx trip on manual operator action
3.0	Manual action to start SI
5.9	Aux. Feedwater throttled
16.0	Ruptured SG isolation
26.0	RCS cooldown begins using one PORV
38.0	Manually begin opening another PORV
44.0	Second PORV fully open
52.5	Open PZR PORV to Depressurize RCS
57.3	Terminate SI
86.3	Break flow terminated

Unit 2 (Model D5 SGs) SGTR Overfill
Sequence of Events

Time (minutes)	Event Description
0.0	Double ended tube rupture initiation
6.2	Rx trip on low RCS pressure
6.7	SI begins on low RCS pressure
11.5	Aux. Feedwater throttled
19.2 min	Ruptured SG isolation
29.2 min	Initiate RCS Cooldown with One PORV
41.2 min	Manually begin opening another PORV
47.2 min	Second PORV fully open
54.3 min	Open PZR PORV to Depressurize RCS
58.7 min	Terminate SI
87.0 min	Break flow terminated

Question 2. The table in response for question 8a. does not exactly match times in Attachment 1. Also, a reactor trip is noted at 20 minutes in the tables but the reference to 30 minutes in the FSAR mark up on page 15-100 has been deleted. Please explain. (ref. Information from the submittal dated April 2 ,1997)

Response: The difference in the times between the two tables is due to a multiplication factor that was applied to the thermal-hydraulic dose inputs provided for the dose analysis. At the time the dose calculations were begun, the overfill analysis had not been finalized. Preliminary thermal-hydraulic inputs were provided such that the dose work could begin in a timely fashion. To assure that these inputs would bound the finalized overfill analysis inputs, a multiplication factor was applied to the time line associated with the preliminary information. This conservatively lengthened the overall time of the event, including the operator action times. The finalized overfill analysis dose inputs were compared to those used in the dose analysis to assure that they were bounded.

The 30 minute time to trip the reactor that was shown in the FSAR was an operator action time assumed in the original SGTR dose analysis. Since that time, Operations has demonsstrated on the simulator that they will trip the reactor within 20 minutes of tube rupture initiation, if the unit has not already tripped on a RPS trip function. The table documenting the timeline for the SGTR is denoted in Table 15-49 of the FSAR which will be updated upon completion of the final dose calculations. The time noted on page 15-100 was redundant information.

Question 3. Do we have plant specific approval to use RETRAN analysis for overfill? When was approval granted?

Response: The use of the RETRAN computer code for FSAR Chapter 15 analyses has been approved in Topical Reports DPC-NE-3000, 3001, and 3002.(ref. SER dated 12/28/95) Duke Power has previously performed FSAR Chapter 15 SGTR analyses to determine thermal-hydraulic dose inputs. The single failure assumed in these analyses was a failed open steam line PORV. The primary differences between how the previous SGTR analyses were performed and how an overfill analysis is performed are the initial conditions, boundary conditions, and single failure assumed for each. The overall evolution and mitigation of the SGTR event is essentially the same for both concerns, with break flow terminating at different times due to the differences in initial conditions, boundary

conditions, and single failure. Since the initial conditions and boundary conditions are different for SGTR overfill than what was previously approved for the T-H dose inputs analyses, the conservative adjustment of these parameters are based on WCAP-10698 for evaluating SGTR overfill. The limiting single failure assumed was that two steam line PORVs fail closed, which is more limiting than the single failed closed steam line PORV in WCAP-10698 and therefore necessitates that a plant specific analysis be performed. These analyses are performed within the approved limits and conditions defined in the topical reports for Duke Power.