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June 25, 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

Re: Catawba Nuclear Station, Unit 1
Docket No. 50-413
Draft Technical Specifications

Dear Mr. Denton:

In response to your June 20, 1984 letter requesting additional information concerning the Catawba Unit 1 Draft Technical Specifications, please find attached Duke Power's response.

Very truly yours,

Hal B. Tucker

Hal B. Tucker

HBT/ekg

Attachment

cc: Mr. James P. O'Reilly
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Question: 1. <u>Special Test Exception, Reactor Coolant Loops</u>, Section 3/4.10.4 (Page 3/4 10-4)

This technical specification permits plant operation without any reactor coolant pump in operation up to 10% thermal power for startup or physics tests. However, there has been no analysis submitted for Catawba to show that the results of any transient or accident under those operating conditions could meet the acceptance criteria for each event. Justify this specification by either providing the analyses demonstrating acceptable plant operation or referencing other analyses and demonstrating that they bound Catawba.

Response: The Westinghouse safety evaluations written for plants which performed post-TMI natural circulation tests were based on analysis in which protective action did not occur until 10% power. Thus, it is permissable to have the pups off and be less than 10% power for these tests. The plants for which Westinghouse submitted these evaluations were Sequoyah, North Anna, Salem, and Diablo Canyon. After several of these tests were performed the NRC agreed that these evaluations were shown to be generic and were no longer necessary to ensure safe operation during the test. Therefore, it is Duke's position that this specification is satisfactory as written.

The natural circulation test will be performed at 3% power as stated in the FSAR test abstract (Table 14.2.12-2, pages 35 and 36).

Question: 2. Reactor Trip System Instrumentation Setpoints, Table 2.2-1 (Section 2.2-1 page 2-5)

a. The equations in notes 1 and 3 do not correspond to those in Chapter 7 of the FSAR. Resolve this discrepancy.

b. Provide information sufficient to verify the allowable values and trip setpoints for the undervoltage and underfrequency trips.

c. Table 2.1-1 lists various interlock setpoints. Provide the bases for these setpoints. What instrument uncertainties were considered in establishing these setpoints and allowable values?

d. In order to verify the trip setpoints and allowable values, the safety analysis limits listed in the setpoint methodology table need to be compared to those listed in Table 15.0.6-1 of the FSAR. However, the limited information listed in the FSAR does not provide for this comparison. Describe the safety analysis limits for underfrequency, containment pressure high and high-high, steam line pressure low, steam line pressure high negative rate and SG water level high-high (both units). Also, modify the methodology to reflect a value for the safety analysis limit for undervoltage.

e. The safety analysis limit for the SG water level low low is listed as 3.9% in the FSAR and 0% in the methodology. Modify either the FSAR or the methodology to bring these values into agreement. Justify the value chosen.

f. The pressurizer pressure-low reactor trip safety analysis limit in Table 15.0.6-1 of the FSAR (1921 psig) differs from that used in the methodology table (1913 psig).

g. The pressurizer pressure-high reactor trip is listed in the setpoint methodology with a safety limit of 2445 psig and an allowable value of 2399 psig. The FSAR (Table 15.0.6-1) lists this value as 2410 psig and 2396 psig respectively. Resolve this difference.

h. The steam generator low low level allowable value is listed as 15.3% span in the setpoint methodology and as a function of the reactor thermal power in the TS. Resolve this difference. Explain the single value since the SG has a programmed level that varies with power level.

Response: 2. a. The OTΔT and OPΔT equations in Chapter 7 represent the actual trip setpoint while the equations in notes 1 and 3 include additional compensation on the measured ΔT and T_{avg}.

Response: 2. b. The undervoltage and under frequency trip setpoints and allowable values for Catawba are the same as those previously accepted for McGuire. The trip instrumentation is identical at both plants and therefore, the setpoints for McGuire were adopted on Catawba.

A copy of the McGuire setpoint methodology is attached.

It is requested that information which is proprietary to Westinghouse be withheld from public disclosure.

c. There are six permissives identified in Table 2.2-1. Of these, only the P-9 and the P-8 are used in the safety analysis.

The other permissives listed in Table 2.2-1 use setpoints based on considerations for plant operability. The allowable values were derived in a fashion similar to those for the protection system setpoints that use the same channels, i.e. the instrumentation uncertainty of the NIS permissives is the same as the trip functions.

The P-9 permissive is used to demonstrate that deleting the reactor trip on turbine trip below 70% power will not increase the probability of a small break LOCA in compliance with TMI action Item II.3.k.10.

The P-8 is used in the startup of an inactive reactor coolant pump at an incorrect temperature analysis. As stated on FSAR page 15.4-17, the reactor trip is assumed to occur on low coolant loop flow when the power range neutron flux exceeds the P-8 setpoint. The P-8 setpoint is conservatively assumed to be 85% of rated power which corresponds to the nominal setpoint plus 9% for nuclear instrumentation errors. This setpoint is only applicable for N-1 loop operation and is included here for completeness.

The P-10 setpoint is based on the need to transfer RCP motor power from the offsite startup transformer to the generator. Typically, this is done when the plant is self-sustaining at 5-to-8% power. The transfer could cause an undervoltage trip, hence the P-10 interlock to block the trip below 10% power.

The P-6 permissive is set to prevent spurious trips from the source range detectors after the intermediate range detector becomes functional.

The P-7 permissive is used to bypass the high pressurizer level and low pressurizer pressure reactor trips during low power or startup operation. During low power and startup operation, frequent pressurizer surges can occur. In order to minimize spurious, unnecessary reactor trips and still provide core protection, the P-7 permissive is set at 10% power.

Response: 2. c. Continued

The P-11 permissive is set above the low pressurizer pressure SI setpoint to block SI actuation at low pressures during startup and shutdown. Should a SI injection occur at low pressure, the RCS could become overpressurized.

d. See response to question 2.b concerning the underfrequency RCP trip.

Containment pressure high, containment pressure high-high, steam line pressure low, steam line pressure high negative rate and steam generator level high-high signals do not directly actuate a reactor trip. Therefore, it would not be expected that these trips would be included on FSAR Table 15.0.6-1.

The safety analysis limits for the above ESF actuation signals are as stated on the Westinghouse setpoint methodology Table 3-16, except for the undervoltage and the underfrequency setpoints which are discussed in response to question 2.b.

e,f,g. Four differences exist between the Trip Points provided in Table 15.0.6-1 of the FSAR and the Analytical Setpoints used to determine the Reactor Trip Setpoints for Catawba. 2445 psig was used to determine the pressurizer pressure-high reactor trip setpoint though the analysis was performed using 2410 psig. 1913 psig was used to determine the pressurizer pressure-low reactor trip setpoint though the analysis was performed using 1921 psig. These differences are due to addressing increased Barton transmitter uncertainties identified after the analyses were performed. The differences were accounted for in a manner that would not reduce operational margin. The acceptability of this approach is discussed in the following paragraphs. Zero percent was used to determine the steam generator level - low-low reactor trip. The 3.9% value reported in FSAR Table 15.0.6-1 is incorrect. The steam generator level - high-high analytical value of 87.8% used in the setpoint methodology is conservative with respect to the value of 93.6% which was used in the analysis.

1. The only transient for which high pressurizer pressure is assumed to be available is the Loss of Load event. Westinghouse has performed an evaluation for the acceptability of a higher analytical setpoint to address an identified long-term drift concern with Barton transmitters. The results of the evaluation concluded that the design basis for this transient would still be met with insignificant effect on peak pressure and DNBR.

(Continued)

Response· 2. e,f,g. 1. This can be demonstrated for Catawba as illustrated in Figures 15.2.3-1 through 15.2.3-8. Of the four cases analyzed, only case 1 has a DNBR which decreases from its initial value. Increasing the reactor trip setpoint from 2425 psia to 2460 psia would delay the time of reactor trip by less than one second and the minimum DNBR would remain above the limit value. Therefore, Westinghouse has determined that a high pressurizer pressure analytical setpoint of 2460 psia is acceptable.

2. The only transient for which the low pressurizer pressure reactor trip is assumed to be available is the Inadvertent Opening of a Pressurizer Safety or Relief Valve (See Section 15.6.1). Westinghouse has performed an evaluation for the acceptability of a lower analytical setpoint to address an identified long-term drift concern with Barton transmitters. The results of the evaluation concluded that the DNBR design basis for this transient would still be met with insignificant effect on the DNBR.

From Figure 15.6.1-2 it can be seen that reducing the low pressurizer pressure reactor trip from 1936 psia to 1928 psia would result in delaying the reactor trip approximately one second.

As shown in Figure 15.6.1-1, the DNBR is decreasing at such a rate that a one-second delay in reactor trip would still yield acceptable results. Therefore, a pressurizer pressure-low analytical setpoint of 1936 psia is acceptable.

h. This inconsistency has been resolved. See the setpoint study.

Question: 4. Reactor Trip System Instrumentation Setpoint, Bases 2.2.1 (page B 2-3)

a. Undervoltage and Underfrequency (Page B2-7)

Time delays are said to be incorporated in the underfrequency and undervoltage trips to prevent spurious reactor trips. These delays are also said to be set so that the time required for a signal to reach the Reactor Trip breakers shall not exceed 1.2 seconds for the undervoltage or 0.3 seconds for the underfrequency. However, Table 3.3-2 (Reactor Trip System Instrumentation Response Times) shows values of 1.5 and 0.6 seconds for the response times of these protective trips. Which values have been assumed in the accident analyses? What is included in the response time besides the time delay?

b. Intermediate and Source Range, Neutron Flux, (Page B2-5)

We believe the third sentence of this section should be revised to read "...unless manually blocked when P-6 becomes active or automatically blocked when P-10 becomes active." This would make the Bases consistent with FSAR Section 7.2.1.1.2.c. Confirm this belief.

Response: 4. a. The safety analyses assumed 1.5 seconds were available from the time the reactor coolant pump buses lose power until the rods were free to fall. As discussed in FSAR Chapter 7, an intentional time delay is provided to prevent spurious voltage dips from causing a reactor trip. The safety analysis assumed a total of 1.5 seconds response time which includes the intentional delay. The loss of flow underfrequency analysis assumes a 0.6 second total response time including the intentional delay.

b. The Technical Specifications will be revised accordingly.

Question: 5. Reactivity Control Systems, Section 3/4.1 (page 3/4 1-8)

Surveillance Requirement 4.1.2.2a states that "the flow path from the boric acid tanks is greater than or equal to 65°F when it is a required water source." FSAR page 9.3-20 states that all portions of the CVCS which contain 4% boric acid should be maintained at or above 70°F. Clarify this discrepancy.

Response: The applicable portions of the FSAR have been revised to agree with the Technical Specification commitment to maintain the fluid in these lines to $\geq$ 65°F. Note that this is above the temperature at which 4 wt.% boric acid will crystalize (technical data show that the soluability temperature is 59°F for a 4.08 wt.% solution).

Question: 6. Reactor Trip System Instrumentation, Table 3.3-1 (page 3/4 3-2)

a. In Item 18e, the minimum channels operable for interlock P-10 for Mode 1 conflicts with FSAR Sections 7.2.1.1.2.a and b. That is, when coming down in power it takes 3 out of 4 P-10 channels to reinstate the intermediate range high neutron flux trip and the low power range neutron flux trip. This item shows 2 out of 4. Explain this discrepancy and modify the TS, including the appropriate action statement as needed.

b. FSAR Section 7.2.1.1.3 notes that a blocking action of reactor coolant flow-low will occur when 3 of 4 neutron flux power range signals are below the P-8 setpoint. Item 18c lists the interlock as 2 out of 4. Explain this discrepancy and modify the TS, including the appropriate action statements, as needed.

Response: The Catawba Technical Specification reflects the requirements specified in the standard Technical Specifications for the P-10 interlock. The description in the Catawba FSAR is correct as is.

During the June 13, 1984 meeting, the ICSB reviewer stated that he felt that the FSAR and the Technical Specifications were correct and that no further information was required from the applicant.

Question: 7. Reactor Trip System Instrumentation Response Times, Table 3.3-2 (page 3/4 3-7)

In Item 7, overtemperature ΔT shows a response time of less than or equal to 4 seconds. Table 15.4.1-1 of the FSAR lists a reactor trip at 2 seconds after the setpoint is reached. Table 15.0.6-1 reflects a 6 second delay. The bases section of the TS refers to a 4 second piping delay (pg B2-5). What is the response time? What is the delay time? When will the reactor trip occur? Resolve these questions by providing a sequence of events and modify the TS as necessary.

Response: The overall time response is comprised of several components. Among these are the bypass loop heating, loop transport, RTD response, analog and logic, gripper coil voltage decay, and trip breaker opening.

The FSAR analysis represented the total delay in a conservative fashion. Two seconds are required for loop transport and heating of the manifold; four seconds is the total response time for the other components. Thus, four seconds is the appropriate response requirement for the Technical Specifications.

Question: 8. ESFAS Instrumentation, Table 3.3-3 (page 3/4 3-15)

Items 2.c, 3.b.3, 4.c and 12.c all state that there are four channels of containment - high high. This agrees with Table 7.3.1-1 and Figure 7.2.1-1 of the FSAR. However, pages 15.1-9 and 15.1-12 of the FSAR note that there are three channels. Please resolve this discrepancy.

Response: The Technical Specifications and FSAR Chapter 7 are correct. Pages 15.1-9 and 15.1-12 will be revised to show four channels.

Question: 9. EFSAS Instrumentation Trip Setpoints, Table 3.3-4 (page 3/4 3-27)

a. For Item 5d of Table 3.3-4, provide an analysis to show the adequacy of the Doghouse water level-high values.

b. The setpoint methodology for the values in this table does not provide the trip setpoints or allowable values for loss-of-offsite power, AFW suction pressure-low, RWST level-low, loss of power, or the pressurizer pressure P-11 interlock. Provide this information.

Response: 9. a. The doghouse water level high trip of the main feedwater pumps was added to prevent flooding of equipment necessary to mitigate an intermediate sized feedwater break prior to receipt of a normal feedwater isolation signal. A flooding analysis was performed assuming a main feedwater break between the feedwater isolation valve and a steam generator. Assuming a failure of the check valve at the steam generator in the broken loop it is assumed that the contents of that steam generator are dumped into the doghouse. Auxiliary feedwater pumps start resulting in, at most, an additional 1500 gpm flow into the doghouse until the operator identifies the broken line and trips the proper pump. Assuming the main feedwater pumps are tripped at a level of 12 inches the final level is limited to a value which does not result in flooding equipment necessary to mitigate this break. The setpoint is 11 inches and the required instrument accuracy is specified as $\pm$ 1 inch. The level switch has undergone qualification testing. This testing program included effects of aging, seismic event, operational cycling of the instrument, post-line break environment and internals between recalibration. An analysis of the results will be completed by fuel load and will verify that the instrument uncertainty is less than the 1 inch value used to determine the setpoint. On this basis the present setpoint is acceptable.

b. AFW Suction Pressure-Low

Auxiliary feedwater suction pressure-low interlock transfers suction of the pumps to the assured source of water (nuclear service water) in the event that the non-assured supplies are depleted or otherwise unavailable. A calculation has been performed that analyzes the suction fluid conditions which result assuming the switchover logic is actuated at the present setpoints (including the calculated instrument inaccuracy) and maximum calculated flow rates. The resultant transient fluid conditions were then compared with conditions found acceptable by the pump manufacturer. This setpoint thus provides sufficient protection by providing suction fluid when required and yet minimizes the chance of inadvertent injection of river water into the steam generators.

(Continued)

Response: 9. b. The inaccuracy of the instrumentation was determined by a statistical combination of those individual and/or groups, of error components which are not interactive. The error components which are interactive are grouped together and summed. These sums are then statistically combined with individual and/or other groups of error components with which the sums are not interactive.

The relationship between the error components and the total statistical error allowance for a channel (instrument loop) is:

$$\text{Total Error} = (\text{EA or SA}) + \sqrt{(\text{PMA})^2+(\text{PEA})^2+(\text{SCA+SD})^2+(\text{STE})^2+(\text{SPE})^2+(\text{RCA+RCSA+RD})^2+(\text{RTE})}$$

Where: EA/SA = Environmental Allowance - the effect of a potentially harsh environment, during which the instrument loop must function on component accuracy (typically associated only with a sensor located in the potentially harsh environment). SA - Seismic Allowance will be used if applicable and if greater than EA.

PMA = Process Measurement Accuracy - includes plant variable measurement errors up to but not including the sensor. For this application, percent change in specific volume will be considered.

PEA = Primary Element Accuracy - Ex: Flow element and installation.

SCA = Sensor Calibration Accuracy - Reference accuracy at sensor calibration conditions.

SD = Sensor Drift - The change in the sensor input - output relationship over a period of time at reference conditions.

STE = Sensor Temperature Effects - The effect of an assumed maximum sensor ambient temperature change of 50°F from the sensor normal ambient (calibration) temperature.

SPE = Sensor Pressure Effects - the effect of a static process pressure change.

RCA = Rack Calibration Accuracy - the reference accuracy at rack calibration conditions.

RCSA = Rack Comparator Setting Accuracy - the tolerance on the precision with which a comparator trip value can be set, if applicable.

RTE = Rack Temperature Effects - the effect of rack ambient temperature variations, if applicable.

RD = Rack Drift - the change in the rack input/output relationship over a period of time at reference conditions.

(Continued)

Response: 9. b. Refueling Water Storage Tank (FWST) Level-Low

The FWST level-low setpoint initiates automatic suction switchover of the residual heat removal pumps and gives a control room alarm to alert the operators to manually complete switchover of the remaining ECCS pumps and the containment spray pumps. This level was determined by calculating the total volume of water withdrawn from the FWST during all steps of the switchover (assuming worst single failure) and converting this volume to an incremental tank level. This was added to the calculated tank level necessary to prevent vortexing (assuming all ECCS and containment spray pumps were drawing water at maximum calculated flow rates) and the instrument inaccuracy. This sum then is the instrument setpoint.

The loss-of-offsite power and loss of power ESF trip setpoints were reviewed by the Power Systems Branch and found acceptable as modified by FSAR question Q430.9. A description of these setpoints is contained in FSAR Section 8.3.1.1.2.1.

The P-11 interlock setpoint is discussed in response to question 2.c.

Question: 10. ESF Response Times, Table 3.3-5 (page 3/4 3-35)

a. In Item 3.a.2, feedwater isolation on pressurizer pressure-low is listed with a response time of 7 seconds, but Table 15.6.3-1 of the FSAR lists this as 1 second. Resolve this disparity.

b. Several response times in the table cannot be verified for consistency with the Chapter 15 analyses. Provide additional information giving the origin of these values so that we can verify that the Technical Specification values agree with the accident analyses.

Response: 10. a. FSAR Table 15.6.3-1 (attached) will be clarified to show that the feedwater flow is throttled over twenty seconds using the feedwater flow control valve.

b. The following ESF response times were used in the FSAR analyses:

SIGNAL	TIME (SEC)	ACCIDENT
Containment Pressure - High		
Safety Injection (ECCS)	28(1)	LOCA, SLB
Containment Pressure - High-High		
Containment Spray	45	LOCA, SLB
Containment Air Return and Hydrogen Skimmer Operation	600	LOCA
Pressurizer Pressure - Low		
Safety Injection (ECCS)	25	LOCA, SGTR
Feedwater Isolation	7	SLB
Steamline Pressure - Low		
Safety Injection (ECCS)	12(3)/22(4)	SLB
Reactor Trip	2	SLB
Feedwater Isolation	7	SLB
Steamline Isolation	7	SLB
Steam Generator Water Level High-High		
Turbine Trip	2.5	Excessive Feed
Feedwater Isolation	7	Excessive Feed
Steam Generator Water Level Low-Low		
Motor-Driven Auxiliary Feedwater Pumps	60	LNF/FDBK
Turbine-Driven Aux. Feedwater Pumps	60	LNF/FDBK

LOCA - Loss of Coolant Accident
SLB - Steamline Break
LNF - Loss of Normal Feedwater
FDBK - Feedline Break
SGTR - Steam Generator Tube Rupture
Excessive Feed - Excessive Feedwater Flow

Question: 11. Emergency Core Cooling Systems, Section 3/4.5 (page 3/4 5-1)

Surveillance Requirement 4.5.1.2.c requires verification that each UHI accumulator discharge isolation valve "closes automatically when the water level in the instrument calibration is 93.2 + ____ inches above the working line on the water-filled accumulator." Provide the basis for this requirement and fill in the missing instrument calibration dead band.

Response: The appropriate value is 93.2 ± 2.7 inches.

In addition, the UHI nitrogen-bearing accumulator pressure should be between 1185 and 1285 psig.

See the attached Technical Specification Changes.

UPPER HEAD INJECTION

LIMITING CONDITION FOR OPERATION

3.5.1.2 Each Upper Head Injection Accumulator System shall be OPERABLE with:

a. The discharge isolation valves open,

b. A minimum contained borated water volume of 1807 cubic feet,

c. A boron concentration of between 1900 and 2100 ppm, and

d. The nitrogen-bearing accumulator pressurized to between 1185 and 1285 psig.

APPLICABILITY: MODES 1, 2, and 3.*

ACTION:

a. With the Upper Head Injection Accumulator System inoperable, except as a result of closed isolation valve(s), restore the Upper Head Injection Accumulator System to OPERABLE status within 1 hour or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

b. With the Upper Head Injection Accumulator System inoperable due to the isolation valve(s) being closed, either immediately open the isolation valve(s) or be in HOT STANDBY within 6 hours and be in HOT SHUTDOWN within the next 6 hours.

SURVEILLANCE REQUIREMENTS

4.5.1.2 Each Upper Head Injection Accumulator System shall be demonstrated OPERABLE:

a. At least once per 12 hours by:

1) Verifying the contained borated water level in the surge tank and nitrogen pressure in the accumulators, and

2) Verifying that each accumulator discharge isolation valve is open.

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*Pressurizer pressure above 1900 psig.

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SURVEILLANCE REQUIREMENTS (Continued)

b. At least once per 31 days and within 6 hours after each solution volume increase of greater than or equal to 138.3 gallons by verifying the boron concentration of the solution in the water-filled accumulator;

c. At least once per 18 months by:

   1) Verifying that each accumulator discharge isolation valve closes automatically when the water level ~~in the instrument calibration~~ is 93.2 ± 4 inches above the working line on the water-filled accumulator, and 2.7

   2) Verifying that the total dissolved nitrogen and air in the water-filled accumulator is less than 80 scf per 1800 cubic feet of water (equivalent to 5×10^{-5} pound of nitrogen per pound of water).

d. At least once per 5 years and if the requirements of Specification 4.5.1.2c.2) are not met by replacing the membrane installed between the water-filled and nitrogen-bearing accumulators.

THIS PAGE OPEN PENDING RECEIPT OF INFORMATION FROM THE APPLICANT

Question: 12. Component Cooling Water System, 3/4.7.3 (page 3/4 7-10)

a. No surveillance requirement is specified on valves that are locked, sealed, or otherwise secured in position. Justify why surveillance should not be required for these valves.

b. No Technical Specification is provided on the water temperature. Discuss how the existing Technical Specifications ensure the CCW temperature assumptions in the safety analyses are maintained valid.

c. Section 9.2.2.2 of the FSAR states that with respect to the Component Cooling Water," crossovers and nonessential shared components are isolated on Engineered Safeguards Actuation Signals." Since ESFAS includes more than just safety injection, either modify the FSAR or the Technical Specification surveillance requirement 4.7.3.b which only addresses safety injection.

Response: 12. a. The component cooling water system valves that are locked in position are used to provide maintenance isolation or isolate train crossover connections during normal operation. The expected frequency of operating these valves is very low since they are not routinely operated to change system alignments. Since they are cycled infrequently there is a low probability of misalignment. Additionally, the administrative procedures to obtain a key and log the valve as being out of position, control and document the operation of the valve. Finally, the procedure requiring the valve to be out of position also directs that it be placed into its proper position as a part of returning the system to normal service. These procedures include an independent verification. For these reasons we feel that surveillance of locked valve positions is not necessary.

b. The Component Cooling Water System (KC) is designed to remove heat from various components during all modes of plant operation. As would be expected the heat load of the KC varies depending on plant mode. The most limiting mode from a heat load standpoint is during normal unit cooldown. The component cooling water heat exchanger is sized to handle cooldown with a nominal rise in system temperature. As a result the Component Cooling System is capable of maintaining its outlet temperature of 100°F during all other normal operations. It should also be noted that on a safety injection signal all non-essential components are isolated and the heat load during the injection phase of an accident is well below heat transfer capability of the system. For these reasons a Technical Specification limit on maximum allowable temperature is unnecessary.

c. Surveillance Requirement 4.7.3b.1 has been revised to include all appropriate engineered safeguards signals received by Component Cooling Water System valves.

Question: 13. Nuclear Service Water System, 3/4.7.4 (page 3/4 7-11)

a. Section 9.2.5.1 states that "the nuclear service water system is designed to operate within a specified temperature range." No technical specification is provided on the water temperature. Justify the lack of such a technical specification by demonstrating that other Technical Specifications ensure the nuclear service water temperature will remain within the assumptions in the safety analyses.

b. Same comment and request as 12.a above.

c. According to the FSAR, the nuclear service water system receives engineered safeguards actuation signals--not simply SI signals. As such, is not item 12.c appropriate here also?

Response: 13. a. The maximum Standby Nuclear Service Water Pond (SNSWP) temperature during modes 1, 2, 3 and 4 is specified in LCO 3.7.5.b (presently on page 3/4 7-12).

b. The Nuclear Service Water System valves that are locked in position are used to provide Unit 1 isolation from Unit 2 (before Unit 2 is placed in service), maintenance isolation or isolate train crossover connections during normal operation. The expected frequency of operating these valves is very low since they are not routinely operated to change system alignments. Since they are cycled infrequently there is a low probability of misalignment. Additionally, the administrative procedures to obtain a key and log the valve as being out of position control and document the operation of the valve. Finally, the procedure requiring the valve to be out of position also directs that it be placed into its proper position as a part of returning the system to normal service. These procedures include an independent verification. For these reasons we feel that surveillance of locked valve positions is not necessary.

c. Surveillance Requirement 4.7.4.b.1 has been revised to include all appropriate engineered safeguards signals received by nuclear service water system valves.

Question: 14. Several Technical Specification Sections

The following concern is generic to several other NTOLs with Westinghouse reactors, including the Callaway plant. The staff has accepted use of a generic approach for resolution of this issue and other issues identified in NUREG-1024 (entitled "Technical Specification-Enhancing the Safety Impact," November 1983). Participation in the generic effort would be acceptable to the staff for resolving this concern on Catawba.

The different sections in the Technical Specifications state limits on minimum and/or maximum values of process variables, e.g., temperatures, pressures, flow rates, levels, and volumes. The staff is concerned that (a) the process variable limits are not, in all cases, used in the safety analyses; and that (b) the analyses assumptions do not always start from the specified values in the Technical Specifications after adding an instrumentation error and uncertainty allowance. For example, Section 3.2.5 specifies the maximum T_{ave} to be $\leq$ 595°F during Mode 1. Additionally, the Catawba FSAR states that the temperature error used in the safety analyses is ±6.5°F. Therefore, if the T_{ave} limit in the Technical Specification is $\leq$595°F, then the safety analyses where the higher temperature is more limiting should assume a T_{ave} value of 595° + 6.5 or 601.5°F as an initial condition.

Provide a justification for not assuming in the safety analyses steady state conditions that are consistent with the limits specified in the Technical Specifications after adding a conservative uncertainty margin. Distinguish between the value of the parameter as measured and limited by the Technical Specifications and the value of the parameter as assumed in the safety analyses.

Response: The Accident Analysis assumes event initiation from nominal conditions with allowances for uncertainties such as measurement error and control dead band. These nominal conditions are maintained by automatic control systems such that deviation from the nominal operating points are limited to within the allowance bands. It is not necessary to add to the Technical Specifications restrictions on all process variables used in the Safety Analysis. Where the Technical Specifications do contain restrictions on process variables the specified limiting values are typically actual values, that is either design values or those used in the analysis, without additional allowances for measurement uncertainty. Where it is necessary to consider measurement uncertainty, the Technical Specifications specifically address (with the exception of RCS T_{avg} and Pressurizer Pressure) the manner in which uncertainties are considered. In the case of RCS T_{avg} and Pressurizer Pressure the attached Technical Specification change is hereby proposed to clarify this situation.

(Continued)

Response: 14. All values in the Technical Specifications other than those whose uncertainties are specifically specified whether analytical, design, etc. may be treated as indicated values without regard for instrument uncertainties. This is acceptable because of the relatively small magnitude of typical measurement uncertainties (one-to-two percent of calibrated span) when compared to the conservatisms included in the plant design and safety analysis. These measurement uncertainties are maintained small by conformance to the Operating Quality Assurance Program which includes requirements for controls of Measuring & Test Equipment, Documents, Design, Test and Inspection, and Procedures. Small deviations in tank levels or pressures, pump flow or discharge pressure, etc. resulting from measurement uncertainty are negligible considering the conservatisms upon which the "limiting" values are based.

By the methods described above the operator can compare indicated values (unless allowances for measurement uncertainties are specified) to those values in the Technical Specifications to ensure compliance, thereby eliminating the use of intermediate documents to account for measurement uncertainties.

TABLE 3.2-1

DNB PARAMETERS

PARAMETER	LIMITS Four Loops in Operation
Indicated Reactor Coolant System T_{avg}	$\leq$ 592.5°F
Indicated Pressurizer Pressure	$\geq$ 2220 psig*

*Limit not applicable during either a THERMAL POWER ramp in excess of 5% of RATED THERMAL POWER per minute or a THERMAL POWER step in excess of 10% of RATED THERMAL POWER.

JUN 8 1984

Question: 15. Bases, Pressure/Temperature Limits, Section 3/4.34.9 (page B 3/4 4-7)

For Bases 3/4.4.9, entitled "Pressure/Temperature Limits," discuss how the heatup cooldown rate limits are calculated to keep the plant in conformance with Appendix G.

Also describe how the PORV low temperature overpressure protection setpoint is calculated.

Response: The following paragraphs describe the derivation of the PORV low temperature overpressure protection setpoint and will be added to the Technical Specification Bases.

The maximum allowed PORV setpoint for the Low Temperature Overpressure Protection System (LTOP) is derived by analysis which models the performance of the LTOP System assuming various mass input and heat input transients. Operation with a PORV setpoint less than or equal to the maximum setpoint ensures that Appendix G criteria will not be violated with consideration for a maximum pressure overshoot beyond the PORV setpoint which can occur as a result of time delays in signal processing and valve opening, instrument uncertainties, and signal failure. To ensure that mass and heat input transients more severe than those assumed cannot occur, Technical Specifications require lockout of both safety injection pumps and all but one centrifugal charging pump while in MODES 4, 5 and 6 with the reactor vessel head installed and disallow start of a RCP if secondary temperature is more than 50°F above primary temperature.

The maximum allowed PORV setpoint for the Low Temperature Overpressure Protection System will be updated based on the results of examinations of reactor vessel material irradiation surveillance specimens performed as required by 10 CFR 50, Appendix H and in accordance with the schedule in Table 4.4-5.

DUKE POWER COMPANY
POWER BUILDING
422 SOUTH CHURCH STREET, CHARLOTTE, N. C. 28242

XC → Bob Carpenter

Return to GAC

WILLIAM O. PARKER, JR.
VICE PRESIDENT
STEAM PRODUCTION

TELEPHONE: AREA 704
373-4083

October 8, 1981

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

Re: McGuire Nuclear Station, Unit 1
Docket No. 50-369

Dear Mr. Denton:

As per Mr. Karl Kniel's April 11, 1977 letter, enclosed are:

1. Five copies of Information Related to Reactor Protection System/ Engineered Safety Features Actuation System Setpoint Methodology (Proprietary) (April, 1981)

2. Five copies of Information Related to Reactor Protection System/ Engineered Safety Features Actuation System Setpoint Methodology (Non-Proprietary) (April, 1981)

Also enclosed is:

1. One (1) copy of Westinghouse Application for Withholding, CAW-81-3 (Non-Proprietary).

As this submittal contains information proprietary to Westinghouse Electric Corporation, it is supported by an affidavit signed by Westinghouse, the owners of the information. The affidavit sets forth the basis on which the information may be withheld from public disclosure by the Commission and addresses with specificity the considerations listed in paragraph (b)(4) of Section 2.790 of the Commission's regulations.

Accordingly, it is respectfully requested that the information which is proprietary to Westinghouse be withheld from public disclosure in accordance with 10 CFR Section 2.790 of the Commission's regulations. Correspondence with respect to the proprietary aspects of this application for withholding or the supporting Westinghouse affidavit should reference CAW-81-3 and should be addressed to R. A. Wiesemann, Manager, Regulatory and Legislative Affairs, Westinghouse Electric Corporation, P. O. Box 355, Pittsburgh, Pennsylvania 15230.

8110160303

Very truly yours,

William O. Parker, Jr.

William O. Parker, Jr.

RWO/php
Attachment

cc: (w/o attachment)
Ms. M. J. Graham
NRC Resident Inspector
McGuire Nuclear Station

Mr. James P. O'Reilly, Director
U. S. Nuclear Regulatory Commission
Region II
101 Marietta Street, Suite 3100
Atlanta, Georgia 30303