ATTACHMENT I

PROPOSED TECHNICAL SPECIFICATION AMENDMENT

TABLE 3.7-2

STEAM LINE SAFETY VALVES PER LOOP

VALVE NUMBER	Loop A	Loop B	Loop C	Loop D	LIFT SETTING (± 1%)* (± 1.5%)**	ORIFICE SIZE
1.	SV-20	SV-14	SV-8	SV-2	1175 psig	14.18 $in.^2$
2.	SV-21	SV-15	SV-9	SV-3	1190 psig	14.18 $in.^2$
3.	SV-22	SV-16	SV-10	SV-4	1205 psig	14.18 $in.^2$
4.	SV-23	SV-17	SV-11	SV-5	1220 psig	14.18 $in.^2$
5.	SV-24	SV-18	SV-12	SV-6	1230 psig	14.18 $in.^2$

*The lift setting pressure shall correspond to ambient conditions of the valve at nominal operating temperature and pressure.

** Change Applies to Unit 2 only. Change only applies until first forced outage, Reactor Trip, or refuelling outage.

ATTACHMENT II

TECHNICAL JUSTIFICATION

Technical Justification

Statement of Problem

Setpoint calibration of the subject safety valves is performed with a model 1566 Hydroset. Dresser, the valve manufacturer has revised the Hydroset correction factor from .339 to .352. Catawba Maintenance personnel were informed of this change by the Dresser field service representative.

Recalculating the set-pressure for valves previously set to the old Hydroset correction factor yields set pressures greater than allowed by Technical Specification Table 3.7-2 for four valves: 2SV3-1.3% (loop D), 2SV8-1.2% (loop C), 2SV12-1.1% (loop C), and 2SV14-1.1% (loop B). Since two valves located on loop C are at set pressures greater than those allowed by Table 3.7-2, without a change to TS or temporary waiver of compliance, Catawba Unit 2 is required to reduce power to 65% per TS 3.7.1.

Analysis

The worst case scenario which results in the most significant challenge to the main steam safety valves is a turbine trip from full power. The reactor will trip due to loss of secondary heat sink on whichever trip parameters are exceeded first (overtemperature delta T or pressurizer pressure). The time to reactor trip is relatively insensitive to the safety valve set pressure differences under discussion.

The design pressure of the steam generators is 1185 psig with a 10% allowable overpressure of 1304 psig. Design study MGDS-0176 for McGuire analyzes the issue of safety valve set point drift. This study is also applicable to Catawba after allowances are made for design differences between the stations. This analysis shows that with a 3% set point drift and 5% accumulation, the highest steam generator pressure condition is less than 1304 psig in this worst case scenario. The primary temperature will be slightly greater due to the increased safety valve setpoint however, no primary side design parameters will be exceeded.

The model used for MGDS-0176 is a McGuire model. The major design differences between McGuire and Catawba are: 1) a higher T-ave for Catawba, and 2) two of Catawba's safety relief valves have a setpoint 5 psi higher than the corresponding valves on McGuire. While these differences (T-ave and relief valve setpoints) result in a higher Steam Generator pressure for Catawba than for McGuire, it is still within the 110% design basis. Additionally, this analysis utilized a 3% set point drift whereas the proposed TS change allows only 1.5%, which provides additional margin.

Based upon the new equipment constant, the new calculated setpoints for the valves in question are within 1.1% to 1.3% of their lift settings. This is less than the setpoint drift used in the McGuire analysis (3%), therefore, adequate margin exists to account for the minor differences between the McGuire and Catawba main steam systems.

Therefore, Duke considers that the error in main steam safety valve set points will not result in a condition that would exceed the design limits of the steam generators.

Safety Significance

Since the subject safety valves will continue to function as designed and serve to maintain the secondary side pressure within its design basis limits, while producing a minimal impact on primary side temperature, the consequences of design basis accidents addressed in the accident analysis as stated in Chapter 15 of the FSAR are not adversely affected. Since the incremental change in the subject safety valves' setpoint tolerance does not prevent the valves from performing their design function and does not affect the probability of failure, there is no increase in the probability of design basis accidents addressed in the FSAR.

From a radiological release standpoint, the most conservative accident is a SGTR. Because of the conservative treatment of releases through the safety valves, the subject change in valve setpoints will not increase the calculated duration of atmospheric releases through the valves for a SGTR. Thus, the dose consequence analysis presented in H. B. Tucker's letter dated December 8, 1989 remain bounding, and the subject setpoint change produces no increase in offsite consequences.

Therefore, it is concluded that there is no nuclear safety significance associated with a 3% deviation from the subject safety valves' nominal setpoint.

ATTACHMENT III

DISCUSSION, NO SIGNIFICANT HAZARDS
ANALYSIS, ENVIRONMENTAL IMPACT STATEMENT

DISCUSSION, NO SIGNIFICANT HAZARDS ANALYSIS, ENVIRONMENTAL IMPACT STATEMENT

Discussion

This proposed emergency change to Table 3.7-2 of the Catawba Technical Specifications will change the Lift Setting tolerance from ± 1% to ± 1.5% for the Steam Line Safety Valves on Catawba Unit 2 until the first forced outage, reactor trip, or refueling outage.

On January 26, 1990, during a vendor review of the procedures used to adjust the Steam Line Safety Valves on January 26, 1990, the Dresser Industries representative mentioned that a new equipment constant had been developed for the valves in August of 1989. At this time, the responsible engineer requested an operability evaluation from Design Engineering.

On January 26, 1990 Catawba Unit 1 was entering a refueling outage. On January 27, 1990. the Unit 1 Steam Line Safety Valves were set-up by using the new equipment constant.

The Design Analysis of the Unit 2 Steam Line Safety Valves, which was complete on February 2, 1990, showed that four valves did not meet the ± 1% tolerance given in TS Table 3.7-2. These valves were: 2SV-3 at 1.3%, 2SV-8 at 1.2%, 2SV-12 at 1.1%, and 2SV-14 at 1.1%. After discussion with Station Management, Design, and NRC residents, it was decided that Design Engineering would confirm the information that had been supplied by the Dresser Representative. The information was confirmed by the vendor on February 2, 1990 and valves 2SV-3 on Loop D, 2SV-8 and 2SV-12 on Loop C and 2SV-14 on Loop B were declared inoperable. Operations entered the TS action statement for TS 3.7.1 and began to reduce power to 65% power since two valves in Loop C (2SV-8 and 2SV-12) were inoperable.

A waiver of compliance was requested from the 1% tolerance, per telecon with NRC Staff on February 2, 1990, based on both the small deviation (.3%) from the 1% required tolerance and the Design Analysis (Attachment II, Technical Justification) that there is no nuclear safety significance associated with a 3% deviation from the subject safety valves nominal setpoint. A verbal waiver of compliance was granted on February 2, 1990. A copy of the amended TS page was submitted to the Staff on February 2, 1990. The staff requested that an emergency TS change be submitted on Monday, February 5, 1990.

No Significant Hazards Analysis

10 CFR 50.92 states that a proposed amendment involves no significant hazards considerations if operation in accordance with the Amendment would not:

1) Involve a significant increase in the probability or consequences of an accident previously evaluated; or

2) Create the possibility of a new or different kind of accident from any accident previously evaluated; or

3) Involve a significant reduction in the margin of safety.

The proposed amendment will change the Lift Setting Tolerance for the Steam Line Safety Valves in TS Table 3.7-2 from 1% to 1.5% for Catawba Unit 2 until the first forced outage, reactor trip, or refueling outage.

This proposed change will not involve a significant increase in the probability or consequences of an accident previously evaluated in the FSAR. The steam line safety valves will continue to function as designed. The secondary side pressure will be maintained within its design basis limits, and there will be a minimal impact on primary side temperature. From a radiological release standpoint, the most conservative accident is a SGTR. Because of the conservative treatment of releases through the safety valves, the subject change in valve setpoints will not increase the calculated duration of atmospheric releases through the valves for a SGTR. Thus, the dose consequence analysis presented in H. B. Tucker's letter dated December 8, 1989 remain bounding, and the subject setpoint change produces no increase in offsite consequences. For the above stated reasons the design basis accidents addressed in Chapter 15 of the FSAR are not adversely affected.

The possibility of a new or different accident from any previously evaluated is not created by this change. The requested change in the safety valves lift setting tolerance is small (.5%). This small increase in the setpoint tolerance does not prevent the valves from performing their design function, and does not affect the probability of failure. Since the valves will still perform their function as designed, this amendment will not create the possibility of a new or different kind of accident from any previously evaluated.

This proposed amendment will not involve a significant reduction in the margin of safety. The requested increase in the Lift Setting tolerance to 1.5% is small. This change will not affect the ability of the Steam Line Safety Valves to perform their design function. Design Engineering concluded that there is no nuclear safety significance associated with a 3% deviation from the subject safety valves nominal setpoint, therefore this change will not affect the margin of safety as described in the FSAR.

For all of the above reasons, Duke Power concludes that this proposed amendment does not involve any Significant Hazards Considerations.

Environmental Impact Statement

The proposed Technical Specification change has been reviewed against the criteria of 10 CFR 51.22 for environmental considerations. As shown above, the proposed change does not involve any significant hazards consideration, nor increase the types and amounts of effluents that may be released offsite, nor increase the individual or cumulative occupational radiation exposures. Based on this, the proposed Technical Specification change meets the criteria given in 10 CFR 51.22 (c) (9) for categorical exclusion from the requirement for an Environmental Impact Statement.

ATTACHMENT IV

JUSTIFICATION FOR AN EMERGENCY
CHANGE TO THE TECHNICAL SPECIFICATIONS

JUSTIFICATION FOR AN EMERGENCY CHANGE TO THE TECHNICAL SPECIFICATIONS

This proposed amendment should receive timely review and approval under the Emergency License Provision of 10 CFR 50.91 (a) (5). The Emergency License Provision applies, for reasons explained below, because circumstances out of Duke Power's control made it impossible for this amendment to be submitted subject to the normal 30 day Federal Register public comment notice period (10 CFR 50.91 (a) (2)).

During a vendor review of station procedures used to adjust main steam line safety valves on January 26, 1990, the Dresser Industries representative mentioned that a new equipment constant had been developed by Dresser in August 1989. This was the first notice that Catawba Nuclear Station had received of the change in the equipment constant. Dresser did not officially change the equipment constant. The responsible engineer wrote a Problem Investigation Report, which initiated a Design Engineering evaluation of the operability of the Steam Line Safety Valves. By Friday afternoon, January 26, 1990, the PIR had been processed by the station and was received by Design Engineering. Duke Power procedures normally allow three working days for Design Engineering to make an operability determination. The informal manner in which Dresser notified Duke of the change implied that it was of minor significance, therefore, no change was thought to be warranted in the three working days to assess the situation.

On Monday morning, January 29, 1990, Design Engineering contacted the station and requested the latest test data in order to calculate the operability of the Unit 2 Steam Line Safety Valves. Because of the large amount of data involved, this information was not compiled and supplied to Design Engineering until Wednesday morning, January 31, 1990.

On February 1, 1990 Design Engineering contacted the station and stated that their evaluation would show that four of the valves would not meet the $\pm$ 1% tolerance given in TS 3.7.1. Station Management, Design, and the NRC Residents discussed the situation. It was decided that Design Engineering would seek official confirmation of the information supplied by the Dresser Representative, and an official Operability Statement would be issued on February 2, 1990.

Design issued an operability statement stating that the Unit 2 valves 2SV-3, 2SV-8, 2SV-12, and 2SV-14 were inoperable. Operations entered the action statement for TS 3.7.1 and began coming down to 65% power, because two valves on Loop C were inoperable.

A conference call was held with Compliance, Station Management, NRC Residents, NRC Region II and NRC NRR. The station requested a waiver of compliance from the $\pm$ 1% tolerance on the Lift Setting of the Safety Valves for Unit 2 until the next forced outage, reactor trip, or refueling outage. This was based on the Design Engineering analysis that the small increase in

the lift setting of the valves has no nuclear safety significance. This waiver of compliance was granted. The NRC requested an emergency TS change submittal on February 5, 1990.

Based on the discussion above, this amendment to the TS should be considered under the Emergency License provision of 10 CFR 50.91 (a) (5). Duke Power was not given notice that Dresser Industries had developed a new equipment constant for the Steam Line Safety Valves in August of 1989. When they were informed of the constant change on January 26, 1990, the operability of the valves was evaluated in a timely manner, and a waiver of compliance was requested when it was officially determined that four of the Steam Line Safety Valves were inoperable. Because this change is necessary to continue operation of Catawba Unit 2 at full power, this TS should be considered under the Emergency License Provision.