ATTACHMENT A

TABLE ~~1.11~~-1 (Continued)

RPS LIMITING SAFETY SYSTEM SETTINGS

A Setpoint cannot be set greater than 10% above measured power whenever reactor power is greater than 10% of rated power.

B ~~Inhibited~~ May be bypassed below 10^{-4}% power ~~(if bypass switches are in the "Bypass" position)~~.

C ~~Inhibited~~ May be bypassed below ~~550~~600 psia ~~(if bypass switches are in the "Bypass" position)~~.

D Bypass allowed for containment leak test.

E Inhibited below 15% power.

F For physics testing at power levels less than 10^{-1}% of rated power the low reactor coolant flow and thermal margin/low pressure trips may be bypassed until their reset points are exceeded if automatic bypass removal of 10^{-1}% of rated power is operable.

TABLE 2-1 (Continued)

ENGINEERED SAFETY FEATURES SYSTEM INITIATION INSTRUMENT SETTING LIMITS

	Functional Unit		Channel	Setting Limit
6.	(Continued)	b.	(Continued)	
			ii) Bus 1A4 Side	≥ 3724.08 volts (4.8 ± .5) seconds } Trip
7.	Low Steam Generator Water Level		Auxiliary Feedwater Actuation	≥ 28.2% of wide range tap span
8.	High Steam Generator Delta Pressure		Auxiliary Feedwater Actuation	≤ 119.7 psid

(1) May be bypassed below 1700 psia and is automatically reinstated ~~above~~ prior to exceeding 1700 psia.
(2) May be bypassed below ~~550~~600 psia and is automatically reinstated ~~above 550~~ prior to exceeding 600 psia.
(3) Simultaneous high containment pressure and pressurizer low/low pressure.
(4) Applicable for bus voltage ≤ 2995.2 - 20.8 volts only. (For voltage ≥ (2995.2 - 20.8) volts, time delay shall be > 5.9 seconds.)

TABLE 2-2

INSTRUMENT OPERATING REQUIREMENTS FOR REACTOR PROTECTIVE SYSTEM

No.	Functional Unit	Minimum Operable Channels	Minimum Degree of Redundancy	Permissible Bypass Condition	Test Maintenance and Inoperable Bypass
1	Manual (Trip Buttons)	1	None	None	N/A
2	High Power Level	2[(b)(c)]	1[(c)]	Thermal Power Input Bypassed below 10^{-4}% of Rated Power[(a)(d)]	(e)(f)
3	Thermal Margin/Low Pressurizer Pressure	2[(b)]	1	Below 10^{-4}% of Rated Power[(a)(d)]	(e)(f)
4	High Pressurizer Pressure	2[(b)]	1	None	(e)
5	Low R.C. Flow	2[(b)]	1	Below 10^{-4}% of Rated Power[(a)(d)]	(e)
6	Low Steam Generator Water Level	2/Steam Gen[(b)]	1/Steam Gen	None	(e)
7	Low Steam Generator Pressure	2/Steam Gen[(b)]	1/Steam Gen	Below ~~550~~600 psia[(a)(d)]	(e)
8	Containment High Pressure	2[(b)]	1	During Leak Test	(e)
9	Axial Power Distribution	2[(b)(c)]	1[(c)]	Below 15% of Rated Power[(g)]	(e)(f)
10	High Rate Trip-Wide Range Log Channels	2[(b)]	1	Below 10^{-4}% and above 15% of Rated Power[(a)(g)]	(e)
11	Loss of Load	2[(b)]	1	Below 15% of Rated Power[(g)]	(e)
12	Steam Generator Differential Pressure	2[(b)]	1	None	(e)

a. Bypass automatically removed.

b. If minimum operable channel conditions are reached, one inoperable channel must be placed in the tripped condition within one hour from the time of discovery of loss of operability. The remaining channel may be bypassed for 48 hours and, if an inoperable channel is not returned to operable status within this time frame, a unit shutdown must be initiated. (See Specification (2) and exception associated with the high rate trip-wide range log channel.)

TABLE 2-4

INSTRUMENT OPERATING CONDITIONS FOR ISOLATION FUNCTIONS

No.	Functional Unit		Minimum Operable Channels	Minimum Degree of Redundancy	Permissible Bypass Condition	Test, Maintenance and Inoperable Bypass
1	Containment Isolation					
A	Manual		1	None	None	N/A
B	Containment High Pressure	A	2[a][c]	1	During Leak Test	(f)
		B	2[a][c]	1		
C	Pressurizer	A	2[a][c]	1	Reactor Coolant Pressure Less Than 1700 psia[b]	(f)
		B	2[a][c]	1		
2	Steam Generator Isolation					
A	Manual		1	None	None	N/A
B	Steam Generator Isolation		1	None	None	N/A
	(i) Steam Generator Low Pressure	A	2/Steam Gen [c]	1/Steam Gen	Steam Generator Pressure Less Than ~~550~~600 psia[c]	(f)
		B	2/Steam Gen [c]	1/Steam Gen		
	(ii) Containment High Pressure	A	2[a][c]	1	During Leak Test	(f)
		B	2[a][c]	1		
3	Ventilation Isolation					
A	Manual		1	None	None	N/A
B	Containment High Radiation	A	2[d]	None	If Containment Relief and Purge Valves Are Closed	(f)
		B	2[d]	None		

a A and B circuits each have 4 channels.

b Auto removal of bypass ~~above~~ prior to exceeding 1700 psia.

c Auto removal of bypass ~~above 550~~ prior to exceeding 600 psia.

TABLE 3-2 (continued)

MINIMUM FREQUENCIES FOR CHECKS, CALIBRATIONS AND TESTING OF ENGINEERED SAFETY FEATURES, INSTRUMENTATION AND CONTROLS

Channel Description	Surveillance Function	Frequency	Surveillance Method
18. SIRW Tank Temperature Indication & Alarms	a. Check	D	a. Compare two independent temperature readings.
	b. Test	M	b. Measure temperature of SIRW tank with standard laboratory instruments.
19. Recirculation Actuation Switches	a. Test	R	a. Manual initiation.
20. Recirculation Actuation	a. Test	M	a. Part of test 3(a) using built-in testing systems to initiate STLS.
	b. Test	R	b. Complete automatic test initiated sensor operation.
21. 4.16 KV Emergency Bus Low Voltage (Loss of Voltage and Degraded Voltage)	a. Check	S	a. Verify voltage readings are above alarm initiation on degraded voltage level - supervisory lights "on".
	b. Test	M	b. Undervoltage relay operation simulated one circuit at a time.
	c. Calibrate	R	c. Known voltage applied to sensors and circuit breaker trip actuation logic verified.

Notes:
1) Not required unless pressurizer pressure is above 1700 psia.
2) Not required unless steam generator pressure is above ~~550~~600 psia.

ATTACHMENT B

DISCUSSION, JUSTIFICATION AND NO SIGNIFICANT HAZARDS CONSIDERATIONS

DISCUSSION AND JUSTIFICATION

The Omaha Public Power District (OPPD) proposes to revise the Fort Calhoun Station Unit No. 1 Technical Specifications to increase the maximum bypass pressure for the Steam Generator Low pressure Signal (SGLS) trip setting as contained in Table 1.1 (page 1-10a), Table 2-1 (page 2-64a), Table 2-2 (page 2-67), Table 2-4 (page 2-69), and Table 3-2 (page 3-12).

On January 22, 1993, during a review of calibration procedures for the steam generator pressure loops, it was determined that the SGLS block reset values for all four channels of both steam generators were greater than that allowed by Technical Specifications (TS). The TS allows SGLS to be bypassed (manually) below 550 psia, however, the block reset values were found to range from 562 to 566.5 psia. This issue was reported to the NRC in Licensee Event Report (LER) 93-002 dated February 22, 1993.

Modification MR-FC-85-136 replaced the existing Pressure Indicating Controllers (PICs) during the 1988-89 (Cycle 13) refueling outage. OPPD provided specifications for the PICs requiring a 10 psi tolerance for the block permissive reset span and an adjustable block permissive setpoint. The manufacturer supplied PICs with non-adjustable block permissive setpoints and reset spans of approximately 26 psi. Therefore, due to the non-adjustability of the installed equipment, the technical specification requirement that the SGLS automatically reset at 550 psia cannot be met.

Currently, the Technical Specifications require the SGLS trip to be enabled above 550 psia and allow the SGLS trip to be bypassed below 550 psia. However, the bypass is automatically reset when the RCS pressure is increased to the values of 562 - 566.5 psia as stated above. The proposed change would increase the permissible bypass to 600 psia. This would ensure the automatic enable feature would be below the technical specification limit when instrument drift, process uncertainties and setpoint calibration tolerances are accounted for in the setpoint implementation. The Combustion Engineering Standard Technical Specifications (NUREG-0212 Rev. 2) allow the trip to be inhibited below 600 psia. The SGLS trip setpoint is 500 psia; the revised bypass setpoint would be 100 psia higher, which is consistent with other Combustion Engineering plants.

The 50 psia increase would not change any of the safety analyses for Fort Calhoun since the limiting transients occur at Hot Full Power and Hot Zero Power for the Main Steam Line Break. Technical Specification 2.10.1(1) requires that the reactor shall not be made critical if the average reactor coolant temperature is below 515°F (except during physics tests at less than 10^{-1}% power). The 515°F temperature corresponds to a 777 psia saturation pressure. The difference in saturation temperature between 550 psia and 600 psia is 9.26°F. An approximate 10°F temperature difference does not significantly effect the consequences of an accident in a non-critical mode. Therefore, the increase in the bypass pressure would not have any impact on the safe operation of the Fort Calhoun Station.

ADMINISTRATIVE CHANGES

The heading for Table 1-1 is being revised to correct a typographical error. The heading states this as Table "1.1" and the revision corrects the designation to Table "1-1."

The statements contained in the footnotes (1) and ([illegible] able 2-1 and footnotes b and c on Table 2-4 are being clarified to state that the automatic reset function occurs prior to exceeding the allowable bypass value.

BASIS FOR NO SIGNIFICANT HAZARDS CONSIDERATION

The proposed change to the Technical Specifications for the increase of the Steam Generator Low pressure Signal (SGLS) bypass to 600 psia does not involve a significant hazards consideration because the operation of the Fort Calhoun Station in accordance with this change would not:

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

   The probability of occurrence does not increase since the limiting postulated accident, Main Steam Line Break has been analyzed and is within the design basis of the plant. The consequences, as reported in the Updated Safety Analysis Report, do not increase since the accident is bounded by a hot full power case and would not be considered limiting. The consequences of an accident, when analyzed at 550 psia versus 600 psia, do not increase significantly. The proposed change would still require that the SGLS is enabled prior to the reactor being made critical (except for physics tests, the technical specifications do not require SGLS to be operable during physics testing below 10^{-1}% power). Therefore, the proposed change will not significantly increase the probability or consequences of an accident previously evaluated in the Updated Safety Analysis Report.

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

   It has been determined that a new or different type of accident is not created because no new or different modes of operation are proposed for the plant. The continued use of the Technical Specification administrative controls prevents the possibility of a new or different kind of accident.

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

   These changes will not reduce the margin of safety since the SGLS trip is still automatically enabled prior to the reactor being made critical. The 50 psia increase would not change any of the safety analyses for Fort Calhoun since the limiting transients occur at Hot Full Power and Hot Zero Power for the Main Steam Line Break.

Based on the above considerations, it is OPPD's position that this amendment does not involve a significant hazards consideration as defined in 10 CFR 50.92 and the proposed changes will not result in a condition which significantly alters the impact of the station on the environment. Thus, the proposed changes meet the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9) and pursuant to 10 CFR 51.22(b) no environmental assessment need be prepared.