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TEXAS UTILITIES GENERATING COMPANY

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May 8, 1984

Director of Nuclear Reactor Regulation
Attention: Mr. B. J. Youngblood, Chief
Licensing Branch No. 1
Division of Licensing
U. S. Nuclear Regulatory Commission
Washington, D.C. 20555

SUBJECT: COMANCHE PEAK STEAM ELECTRIC STATION
DOCKET NOS. 50-445 AND 50-446
STEAM GENERATOR LEVEL, LOW-LOW SETPOINTS

REF: B. J. Youngblood to R. J. Gary letter of March 22, 1984, entitled "Resolution of SER Confirmatory Issues (7) and (8) Pertaining to the Comanche Peak Steam Electric Station (Units 1 and 2)"

Dear Mr. Youngblood:

In response to the referenced letter, the Comanche Peak Steam Electric Station (CPSES) FSAR has been revised to correct previous inconsistencies with respect to the Steam Generator Water Level low-low setpoint. The revisions are attached and affect Sections 15.0 and 15.2 of the CPSES FSAR and the response to NRC question 032.103. These revisions will be included in Amendment 49 to the CPSES FSAR.

Please note that the specific values for the Steam Generator Water Level Low-Low Trip Setpoints for Unit 1 and 2 are no longer provided in the response to question 032.103. This response now refers to the Westinghouse setpoint methodology study where these setpoints were calculated (along with the available margins) for review as part of the CPSES technical specifications. Therefore, the discussions relating to margin of safety and the specific trip setpoint values should be deleted from the NRC Staff's evaluation findings with respect to CPSES SER Confirmatory Issue (7) (as enclosed in the NRC Staff letter referenced above).

Sincerely,

J S Marshall for
H. C. Schmidt

DRW/grr
Attachment

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TABLE 15.0-4
(Sheet 2)

Trip Function	Limiting Trip Point Assumed In Analysis	Time Delays (Seconds)	
Low reactor coolant flow (from loop flow detectors)	87% loop flow	1.0	
Undervoltage trip	68% nominal	1.5	
Turbine trip	Not applicable	2.0	
Low-low steam generator level	34.6%* (Unit 1) and 0% (Unit 2) of narrow range level span	2.0	
High steam generator level trip of the feedwater pumps and closure of feedwater system valves, and turbine trip	90% (Unit 1) and 81% (Unit 2) of narrow range level span	2.0	49

* The basis for the Unit 1 limiting setpoint is the Loss of Normal Feedwater analysis. The setpoint used in the Feedline Break analysis was assumed to be < 2%.

If AC power were not lost for this incident the reactor coolant flow
would remain at its normal value and the reactor would trip via the
low-low steam generator level trip with no change in DNBR below the
value at the start of the transient. The reactor coolant pumps would
be manually tripped at some later time to reduce heat addition to the
RCS. The Auxiliary Feedwater System has sufficient capacity, even 5
assuming the worst single failure, to preclude filling the pressurizer
should the pumps not be tripped.

An additional assumption made for the loss of normal feedwater evaluation is that the pressurizer power operated relief valves are assumed to function normally. Operation of the valves maintains peak RCS pressure at or below the actuation setpoint (2350 pounds per square inch absolute (psia)) throughout the transient. If these valves were assumed not to function, the coolant system pressure during the transient would rise to the actuation point of the pressurizer safety valves (2500 psia). The increased RCS pressure, however, results in less expansion of the coolant and hence more margin to the point where water relief from the pressurizer would occur.

Since the two Comanche Peak units will have different steam generators
(see Section 5.4.2), the effect of this difference has been considered
in the analysis. Both types of steam generators are integral preheater
models. The major difference, from the standpoint of accident analysis
for this event, is the slightly higher secondary side mass as a
function of power for the D5 (Unit 2) model. In order to maximize the 5
time until reactor trip on low-low steam generator level occurs and to
insure that the analysis is valid for both units, the initial steam
generator secondary mass was assumed to be 110% of the higher D5 mass.
The low-low steam generator water level trip setpoint was assumed to be
the same mass (lb. mass) for both units (see Table 15.0-4). Note that 49
while a higher secondary mass (larger heat sink) is, in general, a
benefit for primary side heatup events, the assumption of a higher
initial mass results in a delay of approximately 4 seconds until the 5
trip signal occurs, and thus produces a more severe transient.

In addition, all steam generators for both units will be equipped with separate feedwater connections for injection of auxiliary feedwater and main feedwater at low power operation. The major effect of injecting
5 auxiliary feedwater into the upper section of the downcomer is that most of the flow will bypass the preheat region due to the higher resistance to flow in the preheater. This will result in a slight decrease in heat removal capability. However, the auxiliary feedwater injection point is now much closer to the steam generator, resulting in a much smaller volume of hot feedwater which must be purged before the colder auxiliary feed enters the units.

Plant characteristics and initial conditions are further discussed in Section 15.0.3. Plant systems and equipment which are available to mitigate the effects of a loss of normal feedwater accident are discussed in Section 15.0.8 and listed in Table 15.0-6. Normal reactor control systems are not required to function. Power operated pressurizer relief valves are assumed to function in order to present a more limiting transient, as described above. The Reactor Protection System is required to function following a loss of normal feedwater as analyzed here. The Auxiliary Feedwater System is required to deliver a minimum auxiliary feedwater flow rate. No single active failure will prevent operation of any system required to function. A discussion of ATWT considerations is presented in Reference [2].

Results

Figures 15.2-9 and 15.2-10 show the significant plant parameter transients following a loss of normal feedwater.

Following the reactor and turbine trip from full load, the water level in the steam generators will fall due to the reduction of steam generator void fraction and because steam flow through the safety valves con-

3. Safety Function Setpoints

A. Steam Generator Water Level Trip Setpoints

The only high-energy line rupture within the containment for which the steam generator water level provides the primary trip function is a secondary high energy line rupture from an initial high power condition.

In a continuing effort to monitor protection system operability, Westinghouse has reviewed reported LERs for RPS and ESF trip functions, and when available, plant data on functional test results. During this monitoring process, there has been no indication that the trip functions experience any more additional drift than trip functions with setpoints greater than 5% from the top or bottom of the instrument span. As part of the calibration procedures, several points across the calibrated instrument range including the 0% and 100% span values are checked. Inspection of as left/as found data from these checks do not indicate any significant variances from the instrument range intermediate points. In addition, transmitters are purchased such that the upper and lower range limits for the transmitters are greater than the upper and lower range values, (See SAMA Standard PMC 20.1-1973), i.e., the calibrated instrument range is not the design limit for the device. Based on the above, Westinghouse does not believe that functions which have 95% probability setpoints (when including uncertainities using the approved 49

Westinghouse setpoint metholodolgy) within 5%
of the top or bottom of the instrument range 49
will respond any differently than any other
protection function.

Because large steam generator pressure changes are not expected before trip, only the reference leg heatup effects need be considered, and not the effects of system pressure changes.

The basis for determination of the low-low setpoint is the Loss of Normal Feedwater and Feedline Break events. The setpoints were determined by considering the level used in each of the analyses for each unit.

	Unit 1	Unit 2	
Loss of Normal Feedwater	34.6%	0%	
Feedline Break	2.0%	0%	49

For each unit, the setpoint was determined by considering the following errors for feedline break:

- Normal errors (normal channel accuracy, etc.)
- Post-Accident effects on transmitter (radiation and temperature)
- Reference leg effects (post-accident heatup)

Only the normal errors were considered for the Loss of Normal Feedwater event since it does not result in an adverse environment.

The errors were calculated using the approved Westinghouse setpoint methodology based on plant specific data for each unit. For Unit 1 the Loss of Normal Feedwater analysis setpoint plus normal errors was determined to be more
limiting than the Feedline Break analysis 49
setpoint plus errors (normal, post-accident and reference leg effects). For Unit 2, the Feedline Break analysis setpoint resulted in the limiting setpoint.

At CPSES the reference leg effects have been addressed in the determination of the low-low setpoints. Considered a feedline break and peak compartment temperature of 227^{0}F, the error due to reference leg heatup will be approximately 5.2%.

The high-high steam generator water level trip is not required for accident situations that
could cause significant errors in level 36
indication. The setpoint of this trip will remain unchanged.

Steam generator wide range level does not provide any automatic trip functions.

B. Pressurizer Water Level

Pressurizer water level provides no trip function following an accident which results in an adverse environment inside containment.

4. OPERATOR ACTIONS

A graph depicting the level measurement error due to steam generator reference leg heatup is shown in Figure 032.103-1. Error due to system pressure changes are shown in Figures 032.103-2 and 032.103-3 for the steam generator and pressurizer indicated water level respectively. Plots similar to these will be made available to the operators to ensure they are aware of the potential level measurement errors. 36

Furthermore, a remote possibility exists that the fluid in the open reference legs may flash to steam in the depressurized steam generators following a secondary high energy line rupture. Therefore to alert the operator to the possibility of erroneous indications, Westinghouse has recommends that the following caution be inserted in all plant emergency instructions for indicated steam generator water level.

CAUTION

The operator should not rely upon steam generator water level indications in any depressurized steam generators following a high energy line rupture inside containment. This is due to the possibility of reference leg boiling.

The Westinghouse reference Emergency Operating Instructions take the post accident indicated water level errors into account in the specification of the minimum levels required for safety injection terminations. 36

Similar cautions will be included in the CPSES emergency procedures.