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Ref. # 10CFR50.54q

August 26, 1999

U. S. Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, DC 20555

SUBJECT: COMANCHE PEAK STEAM ELECTRIC STATION (CPSES)
DOCKET NOS. 50-445 AND 50-446
RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION
CHANGES TO CPSES EMERGENCY CLASSIFICATION
PROCEDURE (TAC NOS. M5232 AND M5233)

REF: 1) TXU Electric[1] letter logged TXX-99062 from C. L. Terry to the NRC dated March 30, 1999

2) Telephone conversations on July 29 and August 19, 1999, between TXU Electric representatives and Messrs. David H. Jaffe, Ed Fox and Jim O'Brien of NRR

Gentlemen:

The purpose of this letter is to respond to a NRR staff request for additional information concerning TXU Electric's March 1999 submittal for NRC review and approval of changes to the CPSES emergency classification procedure (Reference 1). The request for additional information was discussed in telephone communications between TXU Electric and the NRR staff as documented above (Reference 2).

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[1]TXU Electric was formerly TU Electric. A license amendment request (LAR 99-003) was submitted per TXX-99122, dated May 14, 1999, to revise the company name contained in the CPSES operating license.

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The attachment to this letter provides the additional information. As discussed in the referenced telephone communications, the enclosed information is intended to support and clarify those changes previously requested in TXU Electric's letter TXX-99062.

If there are any questions concerning this information, please contact Mr. Norman Hood, Emergency Planning Manager at (254) 897-5889.

This communication contains the following new commitment which will be completed as noted:

CDF Number:	Commitment:
[27182]	[The specific values for assessing the proposed EAL "Local Rad Reading" will be incorporated into a station procedure (most likely a Chemistry procedure).]

The CDF number is used by TXU Electric for the internal tracking of CPSES commitments.

Sincerely,

C. L. Terry

C. L. Terry

By: Roger D. Walker

Roger D. Walker
Regulatory Affairs Manager

CLW/grj
Attachment

c - Mr. E. W. Merschoff, Region IV
Mr. J. I. Tapia, Region IV (clo)
Ms. G. M. Good, Region IV
Mr. D. H. Jaffe, NRR
Resident Inspectors, CPSES (clo)

NRC Question 1:

Explain the proposed change in logic from EPP-201 Revision 10 (Chart Blocks 3.B and 4.B) which specify emergency escalation based on S/G Tube Rupture > 50 gpm per ABN-103/ 106 to Revision 11 (Chart Block 3.B) which specifies emergency escalation based on S/G Tube Rupture > capacity of available CCP's following SI Actuation.

TXU Response:

Regarding the proposed change from primary-to-secondary tube leakage > 50 gpm per ABN procedures, it has been our experience that the CPSES operators have difficulty in determining this amount of leakage with any consistency, specifically when the tube leakage increases to this magnitude or higher. With respect to Steam Generator Tube Rupture events, observation of crews in the Control Room Simulator during training and past emergency exercises has shown that the operators quickly begin to increase charging (CPSES centrifugal charging pumps have a capacity of approximately 120 gpm each through the normal charging line), reduce letdown, and reduce reactor/turbine power. This action makes obtaining an accurate tube leak estimate difficult if not impossible to achieve. In the past the resulting practice has generally been to over classify the event which we believe is not in the best interest of the public nor the industry. The use of any normal charging lineup was rejected in that these lineups generally bring back the influences of changing reactor/turbine power and introduce undesirable "guess work" into the classification process. Using the proposed logic of "greater than the capacity of available charging pumps following SI actuation" the operators have a very identifiable condition to use as an escalation threshold. Any significant tube rupture event will most likely result in a reactor/turbine trip and Safety Injection (SI) actuation. By changing the escalation logic to a condition that is post SI, we believe that the operators will have a more objective and very observable threshold. This logic is expected to achieve better consistency in determining the appropriate emergency classification of similar events by a variety of emergency response personnel and yet still be satisfied well before a design basis tube rupture event occurs, i.e., the threshold to escalate occurs prior to the reduction in Reactor Coolant System (RCS) inventory becoming a core "uncovery" issue.

NRC Question 2:

What plant locations would be used to obtain the "Local Rad Reading" as proposed for Revision 11 failed fuel indication (Chart Blocks 2.D, 2.F, 3.A, and 4.A), how long would it take to obtain the readings for use by the emergency response organization, and where are the "Local Rad Reading" emergency action level (EAL) values to be documented ?

TXU Response:

The plant physical locations are the same as specified in the referenced letter and are reiterated here:

Room 78 (near Sample Conditioning Rack): If sample flow has been established for normal sampling of the reactor coolant, then a radiation survey measurement could be taken near the RCS Loop 3/8-in. tubing just behind the accessible end of the Sample Conditioning Rack.

Room 78 (near Post Accident Sampling System (PASS) Module): If sample flow has been established to obtain a PASS sample, then a radiation survey measurement could be taken near the 3/8-in. inlet tubing just behind and above the PASS Module.

Room 77B near the Chemical Volume and Control System (CVCS) letdown piping prior to CVCS isolation. If the CVCS has not been isolated then a measurement could be taken near a portion of the 3-in. letdown pipe inside the radioactive penetration area.

> (Note: If the CVCS letdown has not been isolated, the installed radiation monitor with Control Room readout (FFL-*60) should be available and used for indication of failed fuel; however, in the case that this monitor is out-of-service or otherwise unavailable, then a local radiation survey measurement taken in the Room 77B location could provide the needed indication.)

The length of time to obtain a local area radiation survey varies depending on the location of the responding Chemistry and/or Radiation Protection technician and plant conditions. The assumption in this assessment scenario (i.e., response organization desires an estimate of failed fuel based on taking a local area radiation survey reading due to radioactivity released into the reactor coolant) is that the Chemistry technician is in the plant hot laboratory which is inside the Radiation Controlled Area (RCA). During normal operations, the CVCS letdown is in service and a Chemistry or Radiation Protection technician could obtain a local area radiation survey from this line in about five minutes. The reactor coolant sampling system is not normally in service and it would typically take a few minutes to align to get Reactor Coolant System (RCS) flow purged and through the system. Initially, the normal RCS sample system would be isolated and would take approximately 15 minutes to establish flow and allow a sufficient purge to obtain a meaningful radiation survey of the RCS radiological condition. For assessment using the Post Accident Sampling System (PASS) location, there is a 15 minute warmup on the PASS remote operating module (ROM). It would then take 15 to 30 minutes to actually get flow through the system to be able to obtain a meaningful radiation survey reading. It should be noted that the PASS ROM is turned on at the Alert classification to initiate system warm-up as part of the Operations Support Center staffing activities.

Specific values for "Local Rad Reading" will not be provided in the procedure EPP-201 Chart EALs. This is consistent with the existing practice for the EAL "Chemical Analysis", e.g., see the current Revision 10 Blocks 2.D, 2.F, 3.A, 4.A and 5.A. The specific values for assessing the proposed EAL "Local Rad Reading" will be incorporated into a station procedure (probably a Chemistry procedure). Readings that would correlate to/identify in the range of one-percent failed fuel are planned to be incorporated. This is consistent with the current Chart radiation monitor EAL values proposed for deletion.

The installed plant secondary-side radiation monitors (e.g., Main Steam Line and Condenser Offgas) continue to be available to provide readout and/or alarm in the Control Room and other emergency response facilities. This radiation monitor readout/alarm acts as a stimulus to cognizant emergency organization personnel who may then direct, as desired, that a local area radiation survey measurement be taken to assess potential fuel failure per the proposed "Local Rad Reading" EAL.