UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION IV
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January 27, 2004

EA-04-009

Mr. M. R. Blevins, Senior Vice President
and Principal Nuclear Officer
TXU Energy
ATTN: Regulatory Affairs
Comanche Peak Steam Electric Station
P.O. Box 1002
Glen Rose, Texas 76043

SUBJECT: COMANCHE PEAK STEAM ELECTRIC STATION - SPECIAL TEAM INSPECTION REPORT 05000445/2002-09 - PRELIMINARY WHITE FINDING

Dear Mr. Blevins:

This letter discusses a finding that appears to have low to moderate safety significance. As described in the subject inspection report issued on January 9, 2003, the finding involved the failure to identify and correct an indicated flaw in a steam generator tube during Refueling Outage 1RF08. Failure to remove the tube from service resulted in a steam generator tube leak. The finding was characterized as an Apparent Violation pending the determination of its safety significance. As noted in the inspection report, since your staff removed the leaking tube from service, it did not present an immediate safety concern. This finding was assessed based on the best available information, including influential assumptions, using the applicable Significance Determination Process (SDP) and was preliminarily determined to be a White finding. The finding was preliminarily determined to have a low to moderate safety significance because of the inability of the degraded tube to meet its required safety margins, as indicated by the in-situ testing failures and the associated inability to withstand some severe accident conditions, resulting in an increase in the large early release frequency (ΔLERF) for this performance deficiency. We have provided a summary of the Phase 3 Significance Determination as an enclosure to this report.

There were some differences between your safety assessment, as documented in your letters to the NRC dated March 5 and April 9, 2003, and the significance determination performed by the NRC. These differences included your use of thermal-hydraulic models that predicted much lower steam generator tube temperatures during severe accidents than the NRC models predict. This results in much lower tube failure probabilities than in the NRC's analysis. The agency's models are not designed to be overly-conservative, given the current state of knowledge. The staff considers its model results to be the best currently available basis for risk-informing inspection efforts in this technical area. In addition, your analysis treats the flaw as being smaller and stronger than we believe your data actually indicates. We believe that the

NRC model uses your data in a more objective manner. Finally, in the last of the models you offered for consideration (the fully linked Level 1 and 2 logic model treatment) there appears to be a problem with the truncation level used in the analysis. If corrected, we believe that your model may well support a risk conclusion similar to the conclusion of the NRC's risk analysis.

The finding is also an apparent violation of NRC requirements, as discussed in Section 02.02 of the inspection report, and is being considered for escalated enforcement action in accordance with the "General Statement of Policy and Procedure for NRC Enforcement Actions" (Enforcement Policy), NUREG-1600. The current Enforcement Policy is included on the NRC's website at: http://www.nrc.gov/what-we-do/regulatory/enforcement.html

Before we make a final decision on this matter, we are providing you an opportunity (1) to present to the NRC your perspectives on the facts and assumptions used by the NRC to arrive at the finding and its significance at a Regulatory Conference or (2) submit your position on the finding to the NRC in writing. If you request a Regulatory Conference, it should be held within 30 days of the receipt of this letter, and we encourage you to submit supporting documentation at least one week prior to the conference in an effort to make the conference more efficient and effective. If a Regulatory Conference is held, it will be open for public observation. If you decide to submit only a written response, such submittal should be sent to the NRC within 30 days of the receipt of this letter.

Please contact Charles Marschall at 817-860-8185 within 10 business days of the date of the receipt of this letter to notify the NRC of your intentions. If we have not heard from you within 10 days, we will continue with our significance determination and enforcement decision, and you will be advised by separate correspondence of the results of our deliberations on this matter.

Since the NRC has not made a final determination in this matter, no Notice of Violation is being issued for these inspection findings at this time. In addition, please be advised that the number and characterization of apparent violations in the inspection report, referenced in the subject line of this letter, may change as a result of further NRC review.

In accordance with 10 CFR 2.790 of the NRC's "Rules of Practice," a copy of this letter and its enclosures will be available electronically for public inspection in the NRC Public Document Room or from the Publicly Available Records (PARS) component of NRC's document system (ADAMS). ADAMS is accessible from the NRC Web site at http://www.nrc.gov/reading-rm/adams.html (the Public Electronic Reading Room).

Sincerely,

/RA/

Dwight D. Chamberlain, Director
Division of Reactor Safety

Docket: 50-445
License: NPF-87

Enclosure: SDP Phase 3 Summary
cc w/enclosure:
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ENCLOSURE

Significance Determination Process
Phase 3 Summary

A. Overview of Issue

A failure to identify and correct a clearly detectable steam generator tube flaw indication during eddy current examinations in the 2001 refueling outage (1RF08) resulted in the tube remaining in service until it leaked in September of 2002. The tube subsequently failed in-situ testing, indicating that it would not have met design basis accident requirements.

B. **Significance Determination Basis**

1. Reactor Inspection for IE, MS, B cornerstones

a. Phase 1 screening logic, results and assumptions

The team determined that a Significance Determination Process Phase 2 analysis was required because the issue resulted in the failure of a reactor coolant system barrier, specifically a steam generator tube leak.

b. Phase 2 Risk Evaluation

In accordance with Inspection Manual Chapter 0609, Appendix A, the inspectors conducted a Phase 2 estimation using the Risk-Informed Inspection Notebook for Comanche Peak Steam Electric Station, Unit 1, Revision 1. The dominant affected accident sequences are provided in Table b-1. This finding increases the likelihood of an initiating event, specifically, the Steam Generator Tube Rupture; therefore, the initiating event likelyhood was increased by one order of magnitude in accordance with Manual Chapter 0609, Appendix A, Attachment 2.

The Phase 2 estimation resulted in a preliminary WHITE finding. Therefore, the analyst determined that the finding should be evaluated using the Phase 3 process.

TABLE b-1
PHASE 2 DOMINANT ACCIDENT SEQUENCES

Initiating Event	Sequence	Contribution
Steam Generator Tube Rupture (SGTR)	SGTR-EQ1/ISO-SDC	8
	SGTR-EIHP-EQ2/ISO	7
	SGTR-AFW-MKRWST	6
	SGTR-AFW-FB	6
	SGTR-AFW-EIHP	7

c. <u>Phase 3 Analysis</u>

The analysts conducted an assessment of the Comanche Peak steam generator tube degradation that was as independent as was feasible while using the licensee's information to make the assessment specific to Comanche Peak. By reviewing the differences in the analysts and licensee's underlying assumptions of the physical phenomena involved in this analysis, the licensee identified some appropriate sensitivity studies to perform on both analyses. Upon consideration of all results, the analysts concluded that the best estimate point value of the change in large early release frequency (ΔLERF) for this performance deficiency was 5.5 x 10^{-7}/year. This frequency range corresponds to a "white" finding in the significance determination process.

The assessment of risk for this degraded tube was evaluated for the following three types of severe accident sequences:

(1) Core Damage Sequences Initiated by Spontaneous Tube Rupture:

In this case, the *in-situ* test demonstrated that the tube was capable of withstanding pressure differentials in the range encountered during normal operation; therefore, the analysts concluded that there was no increase in risk associated with spontaneous tube rupture sequences.

(2) Core Damage Sequences Initiated by Secondary Depressurization Events that Induce Tube Rupture:

Because the *in-situ* test was not capable of reaching the pressure difference that would be created by a design-basis secondary depressurization event, it was necessary to use information from the eddy current inspections to estimate the burst pressure for the flawed tube. The predicted burst pressure was greater than the differential pressure that would be experienced during a design-basis depressurization event. However, the predicted burst pressure also had a significant amount of uncertainty, leaving some probability that the tube would actually have burst if exposed to the elevated pressure differential.
A probability distribution for the burst pressure was developed based on the *in-situ* test results and the predicted uncertainty in the model. The analysts estimated the induced rupture probability as the fraction of this probability distribution that was below the pressure difference created by the accident sequence. The analysts increased estimated induced rupture probability as a function of time to account for crack growth during the last operating cycle. The analysts used a series of burst probabilities for specific time periods during the last year of operation with the flawed tube.

As shown in Table c-1, the analyst used a frequency of 1 x 10^{-3} per reactor-year for the frequency of a secondary depressurization event, a probability of 0.25 (1 out of 4) that the depressurization event affects the steam generator with the flawed tube, and a conditional probability of 1 x 10^{-2} that the combined secondary depressurization event and tube rupture will result in core damage. The time periods in Table c-1 are based on the licensee's estimated crack growth rate. Summing the risk contributions for all time periods in the year produces a result of 2.35 x 10^{-8} per reactor-year for the increase in core damage frequency.

The analysts adjusted these values to reflect a slower crack growth rate than was estimated by the licensee. The analysts used the 95^{th} percentile crack growth rate (15% of the wall thickness per year) from a Westinghouse database, instead of the licensee's estimate of 27% per year. The slower crack growth rate increased the risk estimate by predicting that the crack had been in a condition susceptible to rupture for a longer period of time before it was discovered. The adjustment in the growth rate resulted in a frequency of 4.2 x 10^{-8}/year. Because the tube rupture and secondary pressure boundary failure create a direct path to the atmosphere, this result is also the ΔLERF.

TABLE c-1
RESULTS OF STAFF'S ANALYSIS FOR SECONDARY DEPRESSURIZATION SEQUENCES

Initiating Event Frequency (per year)	Time Period (fraction of last year)	Probability SG 2 Affected	Burst Probability	CCDP	Core Damage Frequency (per year)
1 x 10^{-3}	3/12	1/4	0	1 x 10^{-2}	0
1 x 10^{-3}	4.5/12	1/4	0.004	1 x 10^{-2}	3.75 x 10^{-9}
1 x 10^{-3}	1.5/12	1/4	0.009	1 x 10^{-2}	2.81 x 10^{-9}
1 x 10^{-3}	1/12	1/4	0.015	1 x 10^{-2}	3.13 x 10^{-9}
1 x 10^{-3}	1/12	1/4	0.024	1 x 10^{-2}	5.00 x 10^{-9}
1 x 10^{-3}	0.9/12	1/4	0.039	1 x 10^{-2}	7.31 x 10^{-9}
1 x 10^{-3}	0.1/12	1/4	0.072	1 x 10^{-2}	1.50 x 10^{-9}
TOTAL					**2.35 x 10^{-8}**

(3) Core Damage Sequences Initiated by Other Phenomena that Induce Tube Rupture by Creating Abnormally High Temperatures in the Tube Material:

The burst pressure predicted in the previous section for the flawed tube is based on its strength at normal operating temperatures. If a core damage accident occurs in a manner that does not depressurize the reactor coolant system before the reactor core melts, physical testing has demonstrated that the hot gases from the core will reach and overheat the steam generator tubes. The increase in temperature substantially reduces the burst pressure of the tube. The pressure difference across the tubes in a steam generator with a depressurized secondary side would be sufficient to rupture this cracked tube at the higher temperatures. Consequently, the portion of the plant's baseline core damage frequency that produces the conditions necessary to rupture the cracked tube becomes an additional contribution to the large early release frequency because of the crack.

The staff's analysis for this contribution starts with the frequency of the plant damage states (PDSs) in the licensee's PRA that have the characteristics necessary to heat the tubes with the reactor coolant system at high pressure. This is multiplied by the probability that the reactor coolant system will not become depressurized before the tube ruptures and by the probability that the secondary side of the steam generator does become depressurized. The product is further reduced by a factor of 0.5 to account for the probability that the tube is in the part of the steam generator tube bundle that heats up most rapidly. The next reduction factor is the fraction of the last year of operation during which the degradation of the tube made it susceptible to failure under these severe conditions. Finally, the licensee's baseline contribution to the large early release frequency is subtracted from the result to produce the incremental change caused by the flaw. An outline of the calculation is presented in Table c-2.

The analysts total risk estimate for the subject finding is the sum of the estimated change in risk for the three types of severe accident sequences documented in the preceding sections. This summation is provided in Table c-3. The result falls into the "white" range for the ΔLERF.

TABLE c-2
ANALYSIS FOR SEVERE ACCIDENT SEQUENCES THAT INDUCE TUBE RUPTURE

Data Used:	**Frequency (per year) or Probability:**
Sum of Relevant PDS Frequencies:	4.0×10^{-5}
Probability of RCS Depressurization during Core Damage Progression:	x 0.5
Probability that Steam Generator Containing the Flaw is Depressurized:	x 0.1
Probability that the Flaw is in Hottest Part of the Tube Bundle:	x 0.5
Fraction of Last Year that the Flaw was Vulnerable to Rupture:	x 0.57
Total Estimated LERF with Flaw:	5.7×10^{-7}
IPE Baseline LERF from Severe Accident Induced Tube Rupture:	$- 5.6 \times 10^{-8}$
TOTAL ESTIMATED ΔLERF:	$\mathbf{5.1 \times 10^{-7}}$

TABLE c-3 **STAFF'S TOTAL RISK ESTIMATE**	
Type of Tube Rupture:	**ΔLERF (per year)**
Spontaneous	0
Induced by Secondary Depressurization Events	$+ 4.2 \times 10^{-8}$
Induced by Core Damage Accidents	$+ 5.1 \times 10^{-7}$
TOTAL ΔLERF	$\mathbf{5.5 \times 10^{-7}}$