April 14, 1997

Mr. C. Lance Terry
TU Electric
Group Vice President, Nuclear
Attn: Regulatory Affairs Department
P. O. Box 1002
Glen Rose, TX 76043

SUBJECT: CORRECTION OF COMANCHE PEAK STEAM ELECTRIC STATION INDIVIDUAL PLANT EXAMINATION INTERNAL EVENTS STAFF EVALUATION REPORT - (TAC NOS. M74397 AND M80982)

Dear Mr. Terry:

On March 10, 1997, our Staff Evaluation Report (SER) of the Comanche Peak Steam Electric Station (CPSES) Individual Plant Examination (IPE) submittal for internal events and internal flood was forwarded to you. Pages 9-13 were missing from the contractor's Technical Evaluation Report (TER) that was attached as Appendix A to the SER. Enclosed are the pages that were inadvertently left out of the TER. We regret any inconvenience this may have caused.

Sincerely,
ORIGINAL SIGNED BY:
Timothy J. Polich, Project Manager
Project Directorate IV-1
Division of Reactor Projects III/IV
Office of Nuclear Reactor Regulation

Docket Nos. 50-445 and 50-446

Enclosure: Technical Evaluation Report
Pages 9-13

cc: See next page

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

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Transients:

- loss of condenser vacuum
- general plant transients, including:
  - -reactor trip
  - -turbine trip
  - -excessive feedwater flow
  - -closure of one MSIV
  - -inadvertent closure of all MSIVs
  - -core power excursion
  - -loss of primary flow
- inadvertent safety injection signal
- main steam line break
- loss of main feedwater
- loss of a DC bus
- loss of safety chilled water (loss of HVAC)
- loss of offsite power
- loss of a non-vital AC bus
- loss of a protection channel
- loss of component cooling water
- loss of station service water (Unit 1)
- loss of instrument air

Other:

- SGTR
- ISLOCA:
  - -accumulator (relief valve; piping failure)
  - -RH suction line (relief valve; piping; pump seal)
  - -excess letdown line (relief valve; piping)
  - -normal letdown line (relief valve; piping)
  - -LPI, cold legs (relief valve; piping; pump seal)
  - -LPI, hot legs (relief valve; piping; pump seal)
  - -SI, cold legs (relief valve; piping; pump seal)
  - -SI, hot legs (relief valve; piping; pump seal)
- Flooding:
  - -auxiliary building (aux bldg., safeguards and el. cntrl bldg.)
  - -turbine bldg. (circulating water and service water systems)

The initiating event analysis seems to be comprehensive and encompasses most commonly encountered initiating events. Furthermore, it was based on plant specific analyses.

2.2.1.2 Event Trees

The IPE developed 19 event trees plus 5 special event trees to model the plant responses to internal initiating events. Almost every initiating event has an event tree developed for it. The exceptions are flooding, which uses existing event trees, and ISLOCA, which either results in core damage directly, if it is outside the containment, or modifies the in-containment LOCA frequency otherwise, and uses the appropriate LOCA event trees. The five special event trees are for ATWS, and various transient induced LOCAs. They are the transient induced LOCA, the induced small seal LOCA, the induced LOCA 0.6"-2", and the induced LOCA >2" event trees.

Note that the Reactor Vessel Rupture Event tree (or as it is called in the IPE, "the Excessive LOCA event tree", since there is a possibility of multiple pipe failures too) is developed only for the purpose of defining plant damage states. This event is assumed to lead directly to core damage.

The event trees are functional. The mission time used in the core damage analysis was 24 hours.

The event tree end states, other than transfers to other event trees), are divided into the two possible core conditions: stable or damaged (further subdivided into plant damage states). A stable core status is defined to include the hot standby or any other condition where heat removal from the core could continue for an extended period of time, with no requirement for additional systems to operate.

Core damage seems to mean either core uncovery (for most initiators) or exceedence of a critical peak cladding temperature (for large LOCAs).

Success criteria were based on a best estimate plant response; when necessary, MAAP runs or thermal-hydraulic analyses were used to validate the success criteria.

Success criteria for the injection phase of a large LOCA specify one charging or SI pump, 2 accumulators on the unbroken loops and either RH pump. Switchover from cold leg to hot leg recirculation is conservatively assumed to be required, at 18 hours into the accident. Only in case of large LOCA is the reactor trip not required; for all other initiators it is required. For medium LOCA, success criteria specify one charging or SI pump and 2 accumulators on the unbroken loops. High head recirculation (either via charging or SI pumps) is used. Hot leg recirculation switchover is required. In medium or small LOCAs, credit is given for the operators extending the injection phase by throttling the pump flow or replenishing the RWST, such that recirculation may not be needed within the mission time. Small LOCA is defined such that secondary cooling is not needed, and successful injection via charging or SI pumps can be accomplished. In case of a very small LOCA, secondary heat removal via AFW is also needed for success.

In the ATWS event tree, emergency boration is required for success, unless the operators are able to shutdown the reactor by manual rod insertion or local breaker disconnect. Various combinations of number of PORVS, AFW flow rate, success or failure of manual rod insertion and time in core cycle (for moderator temperature coefficient) are considered in order to determine whether or not the RCS pressure will exceed the critical limit of 3200 psia. While the AMSAC system automatically trips the turbine and starts AFW, the operator is credited with performing these actions manually should AMSAC fail. A turbine trip and either half or full AFW flow (meaning all three pumps), depending on other parameters, is required to avoid core damage. Both electrical and mechanical ATWS events are considered in the event tree.

For transients, credit is given for operator action to restore main feedwater (which would have tripped off) in case of failure of auxiliary feedwater. No credit seems to be given for depressurizing and operating only with the condensate pumps. Also, in case of some LOCAs, no credit seems to be given to depressurizing the RCS and using only the RHR pumps, in case the SI or the charging pumps are unavailable (which, in this plant would be a much rarer occurrence than at other plants, due to the number of pumps which can be used for high pressure injection).

In case of a loss of all RCP seal injection and cooling (e.g., via a loss of SW event) some time is available for the operators to take corrective actions (e.g., align alternate cooling for the CCPs via firewater or demineralized water, or via SW cross-connect from the other unit). If unsuccessful, the resulting seal LOCA can be either a large or a small seal LOCA, with split fractions and flow rates given in the submittal or the RAI responses.

Containment heat removal is not required for success in the Level 1 analysis, based on "realistic modeling". Therefore, containment heat removal systems are not considered in the Level 1 event trees or fault trees.

For ISLOCAs, it is not clear if credit is given to isolation, as the RAI responses were not complete in this regard. Consideration is given to the fact that some ISLOCA scenarios spill inside the containment, thus contributing to the regular LOCA frequencies. Also, the resulting flow rate seems to have been considered, and existing event trees used for the analysis, with certain functions and equipment disabled.

Overall, Comanche Peak success criteria and event tree modeling seem to be reasonable and in line with most other IPEs or PRAs for similar plants.

2.2.1.3 Systems Analysis

A total of 15 systems/functions are described in Section 3.2 of the Submittal. Included are descriptions of the following systems: component cooling water, auxiliary feedwater, residual heat removal, station service water, containment spray, chemical and volume control, reactor coolant, safety injection, condensate and feedwater, main steam, circulating water, reactor protection, electric power, instrument air and safety chilled water (HVAC) systems.

Each system description includes a discussion of the functions of the system and the relationship of the various subsystems in fulfilling these functions, maintenance and surveillance activities performed on the system, system actuation signals, principal operator interfaces and system success criteria.

Also included for many systems are simplified schematics that show major equipment items and important flow and configuration information.

2.2.1.4 System Dependencies

The IPE addressed and considered the following types of dependencies: shared component, instrumentation and control, isolation, motive power, direct equipment cooling, areas requiring HVAC, operator actions and environmental effects. HVAC is an important system at this plant, as many systems require it (see Section 1.2 for the list of HVAC-dependent systems), among them ECCS systems, CCW, and emergency power. The TDAFW pump being a notable exception. Therefore, HVAC failure is modeled both as an initiating event and as a subsequent failure.

Table 3.2.3-1 of the submittal contains the support-on-support and frontline-on-support dependency matrix.

2.2.2 Quantitative Process

2.2.2.1 Quantification of Accident Sequence Frequencies

The IPE used a small event tree/large fault tree approach with fault tree linking to quantify core damage sequences. The event trees are functional.

The system and event tree models were developed and quantified within the CAFTA software.

The functional sequences which resulted had cutsets with a truncation limit of 1.E-9/yr. The IPE analysis took credit for various recovery activities, including the recovery of offsite power.

Convolution methodology, including adjustments in the core uncovery times for later failures, was used in the IPE, for evaluating LOOP and SBO cutsets.

The IPE data used for non-recovery of offsite power are lower (by a factor of 2-5) than the average industry data cited in an Electric Power Research Institute (EPRI)-sponsored study (NSAC 147).

The optimistic offsite power recovery factors, along with taking credit for diesel generator repair and AFW pump repair, will have a significant impact on the results. This is especially true since the LOOP and SBO sequences are already a dominant contributor to the CDF (29%).

2.2.2.2 Point Estimates and Uncertainty/Sensitivity Analyses

Mean values were used for the point estimate initiator frequencies.

No uncertainty analysis was performed. Limited sensitivity analyses were provided as part of RAI responses. Limited importance analysis results (for systems) were provided as part of the DHR vulnerability issue.

2.2.2.3 Use of Plant Specific Data

No plant specific data was used, except, apparently, test/maintenance data for pumps and valves. Technical specifications were used for some test/maintenance unavailabilities, along with some computerized plant data.

2.2.2.4 Use of Generic Data

The primary source for the generic data was the PLG database (PLG-0500). Some data comes from IDCOR IPEM 2.4-1A, the Crystal River 3 PRA, the Oconee PRA and expert opinion. Both demand and time dependent failures were considered. Table 2 shows a comparison of failure data used in the CPSES study with that used in the NUREG-1150 study, documented in NUREG/CR-4550.

The failure data used shows a generally good agreement with the NUREG/CR-4550 data. The TDAFW run failure data are somewhat lower, but are within reach, as opposed to most other IPEs whose TDAFW pump run failure rate is about 2 orders of magnitude lower than that in NUREG/CR-4550. The air compressor failure rate is significantly lower than expected, which may have an impact on the results due to high reliance on compressed air at this plant. This is also true of the apparent lack of consideration of air compressor common cause failure (see a next section).

Table 2. Comparison of Failure Data

Component	CPSES	4550
Turbine Driven Pump		
fail to start	3.3E-2	3.0E-2
fail to run	1.0E-3	5.0E-3
Motor Driven Pump, operating		
fail to start	2.4E-3	3.0E-3
fail to run	3.4E-5	3.0E-5
Motor Driven Pump, standby		
fail to start	3.3E-3	3.0E-3
fail to run	3.4E-5	3.0E-5
Air Compressor		
fail to start	3.3E-3	8.0E-2
fail to run	9.8E-5	2.0E-4
Battery Charger Failure	1.9E-5	1.0E-6
Battery failure		
fail on demand	4.8E-4	---
fail during op.	7.5E-7	1.E-6
Circuit Breaker (> =480V)		
fail to remain closed-spur open	8.3E-7	1.0E-6
fail to close	1.6E-3	3.0E-3
fail to open	6.5E-4	3.0E-3
AC Bus Fault	5.0E-7	1.0E-7
Check Valve		
fail to open	2.7E-4	1.0E-4
fail to close	2.7E-4	1.0E-3
MOV Fail on Demand	4.3E-3	3.0E-3
transfer open	9.3E-8	5.0E-7
transfer closed	9.3E-8	1.0E-7
Pressurizer PORV		
fail to open	4.3E-3	2.0E-3
fail to close	2.5E-2	2.0E-3
Emergency Diesel Generator		
fail to start	3.8E-2	3.0E-2
fail to run	2.5E-3	2.0E-3

Notes: (1) 4550 are mean values taken from NUREG/CR-4550, i.e. from the NUREG-1150 study of five U.S. nuclear power plants. (2) Demand failures are probabilities per demand. Failures to run or operate are frequencies expressed in number of failures per hour.