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

November 12, 1996

SUBJECT: REVIEW OF PRELIMINARY ACCIDENT SEQUENCE PRECURSOR ANALYSIS OF EVENT AT COMANCHE PEAK STEAM ELECTRIC STATION, UNIT 1

Dear Mr. Terry:

Enclosed for your information is a copy of the final Accident Sequence Precursor analysis of the operational event at Comanche Peak Steam Electric Station, Unit 1, reported in Licensee Event Report Nos. 445/95-003 and -004. This final analysis (Enclosure 1) was prepared by our contractor at the Oak Ridge National Laboratory, based on review and evaluation of your comments on the preliminary analysis and comments received from the NRC staff and from our independent contractor, Sandia National Laboratories. Enclosure 2 contains our responses to your specific comments. Our review of your comments employed the criteria contained in the material which accompanied the preliminary analysis. The results of the final analysis indicate that this event is a precursor for 1995.

Please contact me at (301) 415-1038 if you have any questions regarding the enclosures. We recognize and appreciate the effort expended by you and your staff in reviewing and providing comments on the preliminary analysis.

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 No. 50-445

Enclosures: 1. Final analysis of Licensee Event Reports 445/95-003 and 004
2. Response to licensee comments

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Mr. C. Lance Terry
TU Electric Company

Comanche Peak, Units 1 and 2

cc:
Senior Resident Inspector
U.S. Nuclear Regulatory Commission
P. O. Box 1029
Granbury, TX 76048

Regional Administrator, Region IV
U.S. Nuclear Regulatory Commission
611 Ryan Plaza Drive, Suite 400
Arlington, TX 76011

Mrs. Juanita Ellis, President
Citizens Association for Sound Energy
1426 South Polk
Dallas, TX 75224

Mr. Roger D. Walker, Manager
Regulatory Affairs for Nuclear
Engineering Organization
Texas Utilities Electric Company
1601 Bryan Street, 12th Floor
Dallas, TX 75201-3411

Texas Utilities Electric Company
c/o Bethesda Licensing
3 Metro Center, Suite 610
Bethesda, MD 20814

George L. Edgar, Esq.
Morgan, Lewis & Bockius
1800 M Street, N.W.
Washington, DC 20036-5869

Honorable Dale McPherson
County Judge
P. O. Box 851
Glen Rose, TX 76043

Office of the Governor
ATTN: Susan Rieff, Director
Environmental Policy
P. O. Box 12428
Austin, TX 78711

Arthur C. Tate, Director
Division of Compliance & Inspection
Bureau of Radiation Control
Texas Department of Health
1100 West 49th Street
Austin, TX 78756-3189

LER No. 445/95-003, -004

Event Description: Reactor trip, Auxiliary Feedwater (AFW) pump trip, second AFW pump unavailable

Date of Event: June 11, 1995

Plant: Comanche Peak 1

Event Summary

While at 100% power on June 11, 1995, Comanche Peak 1 experienced a control power supply failure resulting in both main feedwater pumps (MFPs) tripping and operators subsequently initiating an anticipatory reactor trip. Flow from one of two motor-driven auxiliary feedwater pumps (MDAFWP) was initially unavailable and the turbine-driven auxiliary feedwater pump (TDAFWP) started on low-low steam generator level but tripped on overspeed. The conditional core damage probability (CCDP) estimated for this event is 6.5×10^{-5}.

Event Description

While at 100% power on June 11, 1995, Comanche Peak 1 experienced a control power supply failure resulting in both MFPs tripping and operators subsequently initiating an anticipatory reactor trip. Slave relay testing was under way when a non-safety related inverter transferred from its normal inverter ac power supply to its alternate power supply. The alternate ac power supply was deenergized as required by the test procedure at the time, so associated loads were deenergized. The specific cause of the transfer is not certain but it may have been caused by an electrical transient in a static transfer switch control circuit. Loss of the power supply caused a spurious "MFP oil pressure low" signal when auxiliary relays in pump supervisory instrumentation deenergized and actuated. This change caused the condensate pumps to trip; loss of the condensate pumps caused both MFPs to trip. Operators then initiated a manual reactor trip in anticipation of an automatic one.

The MFP trips caused an auto-actuation of the MDAFWPs. MDAFWP 1-02 (Train B) started and supplied water to steam generators (SGs) 3 and 4 (Fig. 1). MDAFWP 1-01 (Train A) was aligned to its test header at the time and was not immediately available to supply water to the SGs. The TDAFWP started on low-low SG level but tripped on overspeed, caused by a failure of the governor valve to control turbine speed. The governor valve stem was found to be corroded and binding against the valve packing. Operators realigned MDAFWP 1-01 from the test header to its normal configuration, and the pump supplied cooling to SGs 1 and 2 within about 8 min.

Additional Event-Related Information

The licensee event report (LER) provided additional information concerning the thermal-hydraulic effects of having only one AFW pump available immediately after a plant trip. Plant safety analyses assume for a "Loss of Normal Feedwater Flow" transient that the TDAFWP or both MDAFWPs provide a flow rate of at least 860 gpm to the SGs. During this transient, only one MDAFWP was initially available, providing a reduced flow rate to the SGs. However, the LER indicated that the reduced flow rate was adequate to remove plant decay heat from the SGs because of the early manual trip of the reactor and because initial water levels in the SGs were greater than the assumption used in the FSAR analysis. Because sufficient heat removal capability was available, the thermal expansion of the reactor coolant system inventory did not fill the pressurizer completely.

ENCLOSURE 1

Modeling Assumptions

This event was modeled as a reactor trip with the TDAFWP failed and flow from MDAFWP 1-01 initially unavailable. Basic event AFW-TDP-FC-1C was set to "TRUE" (failed). (Table 1 provides a description of the basic event names.) It was assumed that if the remaining AFW pump had failed, operators would have attempted to recover the system by realigning MDAFWP 1-01 (as they did). Recovery of MDAFWP 1-01 was incorporated into the models using the methodology described in Reference 4. This methodology suggests a nonrecovery probability of 0.1 when "[f]ailure appeared recoverable in the required period from the control room, but recovery was not routine or involved substantial stress." A similar nonrecovery value was estimated by assuming that nonrecovery as a function of time was lognormally distributed with a median response time of 8 min and a recovery window of 30 min. Assuming a burdened-recovery error factor of 6.4, the probability of nonrecovery within 30 min is approximately 0.1, which is the same value as obtained using Ref. 4. Consequently, the nonrecovery probability for MDAFWP 1-01 was incorporated by setting the probability for event AFW-MDP-FC-1A equal to 0.1. In addition, because AFW is required without delay during ATWS sequences, a new event, AFW-MDP-FC-AA, with a nonrecovery probability of 1.0 was substituted for AFW-MDP-FC-1A in the ATWS model. Because it was assumed that the entire 30 min would be dedicated to recovery of AFW-MDP-FC-1A, the system nonrecovery, AFW-XHE-NOREC, was set to 1.0.

Because main feedwater apparently could not have been recovered without correcting the inverter problem, restarting the condensate system, and restoring a feedwater pump to service, the feedwater system was assumed not to be recoverable (MFW-XHE-NOREC = "TRUE").

The failures in this event increase the potential significance of failure to trip/ATWS sequences. To model potential reactor trip failures more accurately, the reactor trip model was modified (as shown in Fig. 2) to account for recoverable versus nonrecoverable RPS failures.

The event trees for Comanche Peak assume that conditions requiring a reactor trip will first result in an automatic reactor trip demand and, if the automatic trip fails, a manual reactor trip demand. During this event, once operators recognized that a loss of main feedwater flow had occurred, they initiated a manual reactor trip. Because of the operators' quick response, consideration was given to the potential impacts of the early reactor trip on ATWS sequences. The Comanche Peak Final Safety Analysis Report (FSAR) indicates that 1 to 1½ min may elapse between a loss of feedwater and an automatic reactor trip. The additional 1 min of response time available to operators during postulated ATWS sequences in this event was not believed to materially affect the event sequences or probabilities, and no related model changes were indicated.

Analysis Results

The CCDP estimated for this event is 6.5×10^{-5}. The dominant core damage sequence (sequence 20 on Fig. 3) involves

- a successful reactor trip,
- failure of AFW
- failure of MFW, and
- failure of feed-and-bleed cooling.

The second highest core damage sequence (sequence 21-8 on Figs. 3 and 4) involves

- failure to successfully trip,
- successful control of reactor pressure, and
- failure of AFW.

Definitions and probabilities for selected basic events are shown in Table 1. The conditional probabilities associated with the highest probability sequences are shown in Table 2. Table 3 lists the sequence logic associated with the sequences listed in Table 2. Table 4 describes the system names associated with the dominant sequences. Minimal cut sets associated with the dominant sequences are shown in Table 5.

Acronyms

ac	alternating current
AFW	auxiliary feedwater
ATWS	anticipated transient without scram
CCDP	conditional core damage probability
CST	condensate storage tank
FSAR	final safety analysis report
LER	licensee event report
MDAFWP	motor-driven auxiliary feedwater pump
MFP	main feedwater pump
SG	steam generator
SWS	service water system
TDAFWP	turbine-driven auxiliary feedwater pump

References

1. LER 445/95-003, Rev. 1, "Loss of Both Condensate and Both Feedwater Pumps Due to Failure of Non-Safety Related Inverter Resulted in a Manual Reactor Trip," August 14, 1995.

2. LER 445/95-004, Rev. 1, "Allowed Outage Time Was Exceeded on Turbine-Driven Auxiliary Feedwater Pump Which Tripped on Overspend," September 8, 1995.

3. Texas Utilities Generating Company, *Comanche Peak Steam Electric Station Final Safety Analysis Report*.

4. M. B. Sattison et. al., *Methods Improvements Incorporated into the SAPHIRE ASP Models*, NUREG/CP-0140, Vol. 1, Proceedings of the U.S. Nuclear Regulatory Commission, Twenty-Second Water Reactor Safety Information Meeting, April 1995.

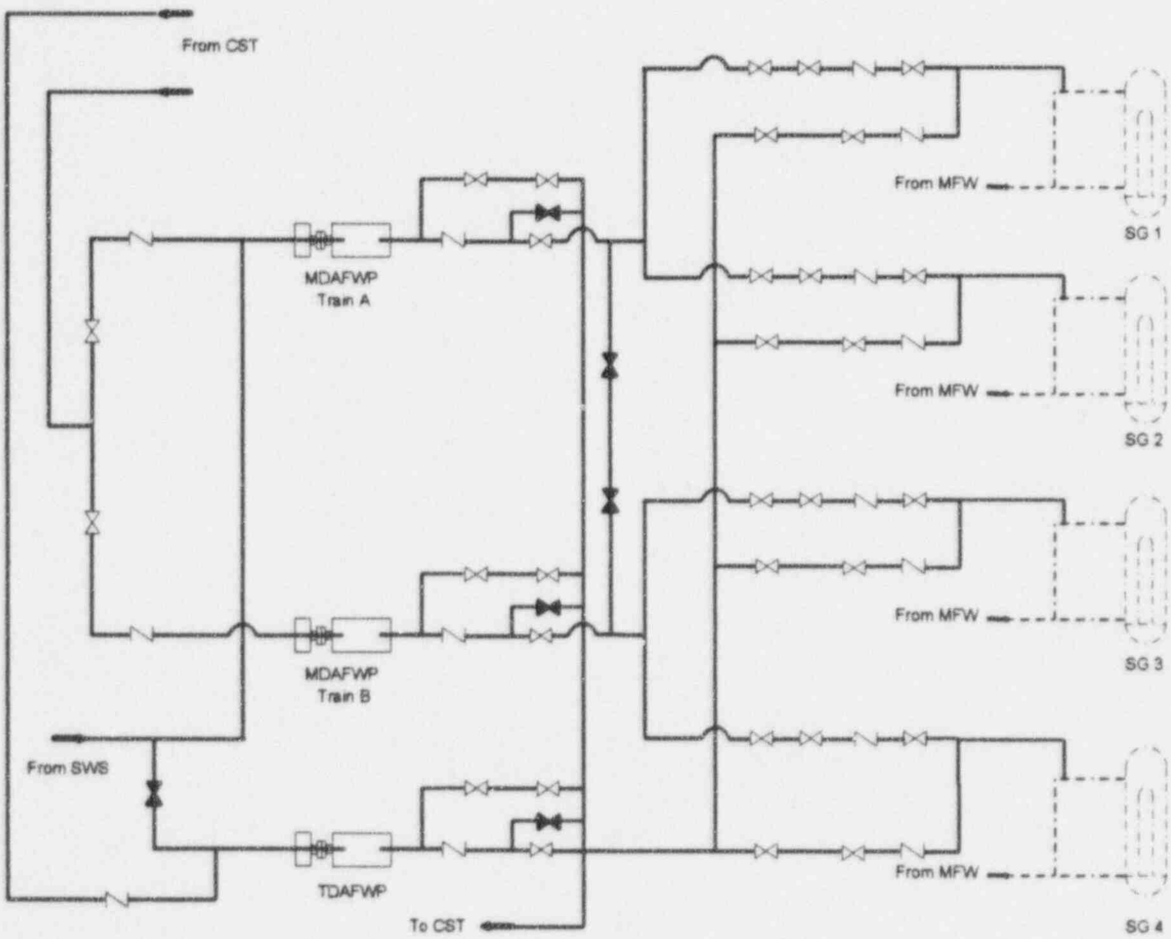

Fig. 1 Auxiliary feedwater system for Comanche Peak (Source: Texas Utilities Electric Co., *Comanche Peak Steam Electric Station Final Safety Analysis Report*)

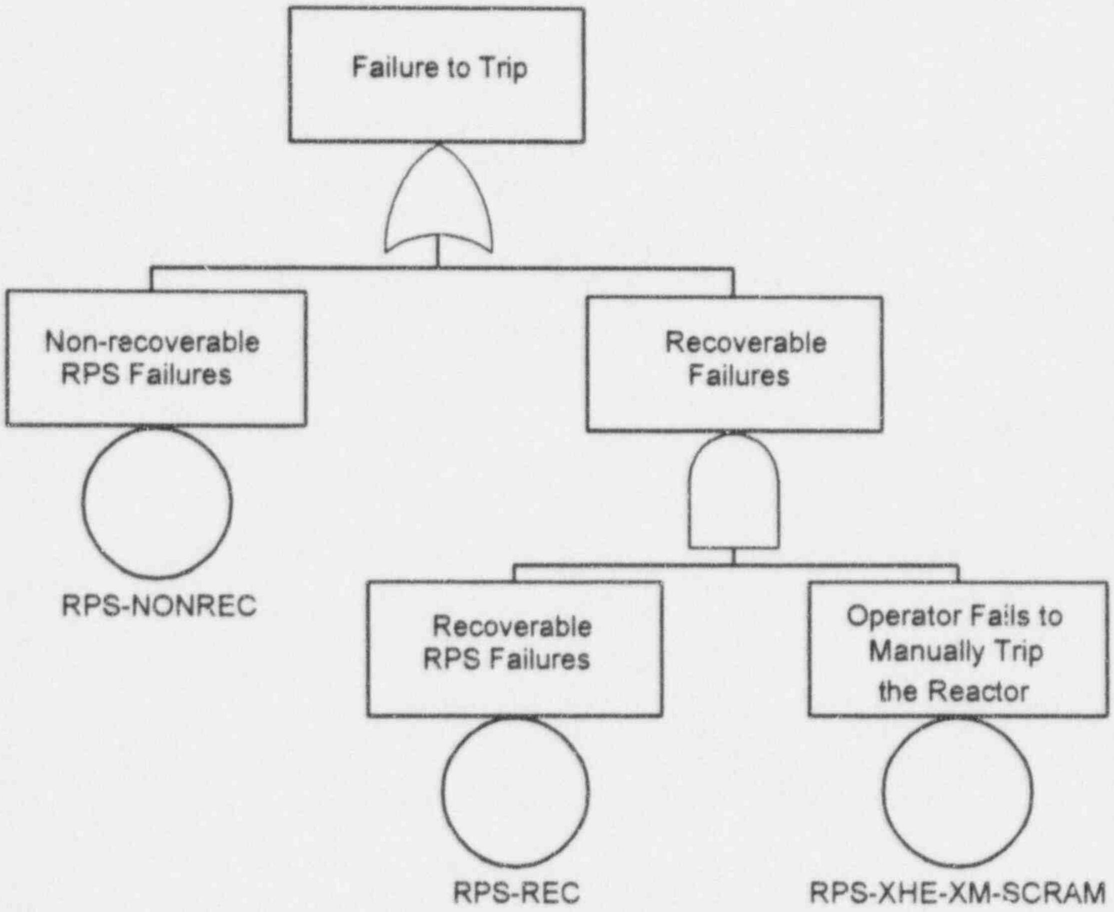

Fig. 2 Fault tree modeling recoverable and nonrecoverable failures for the failure to trip.

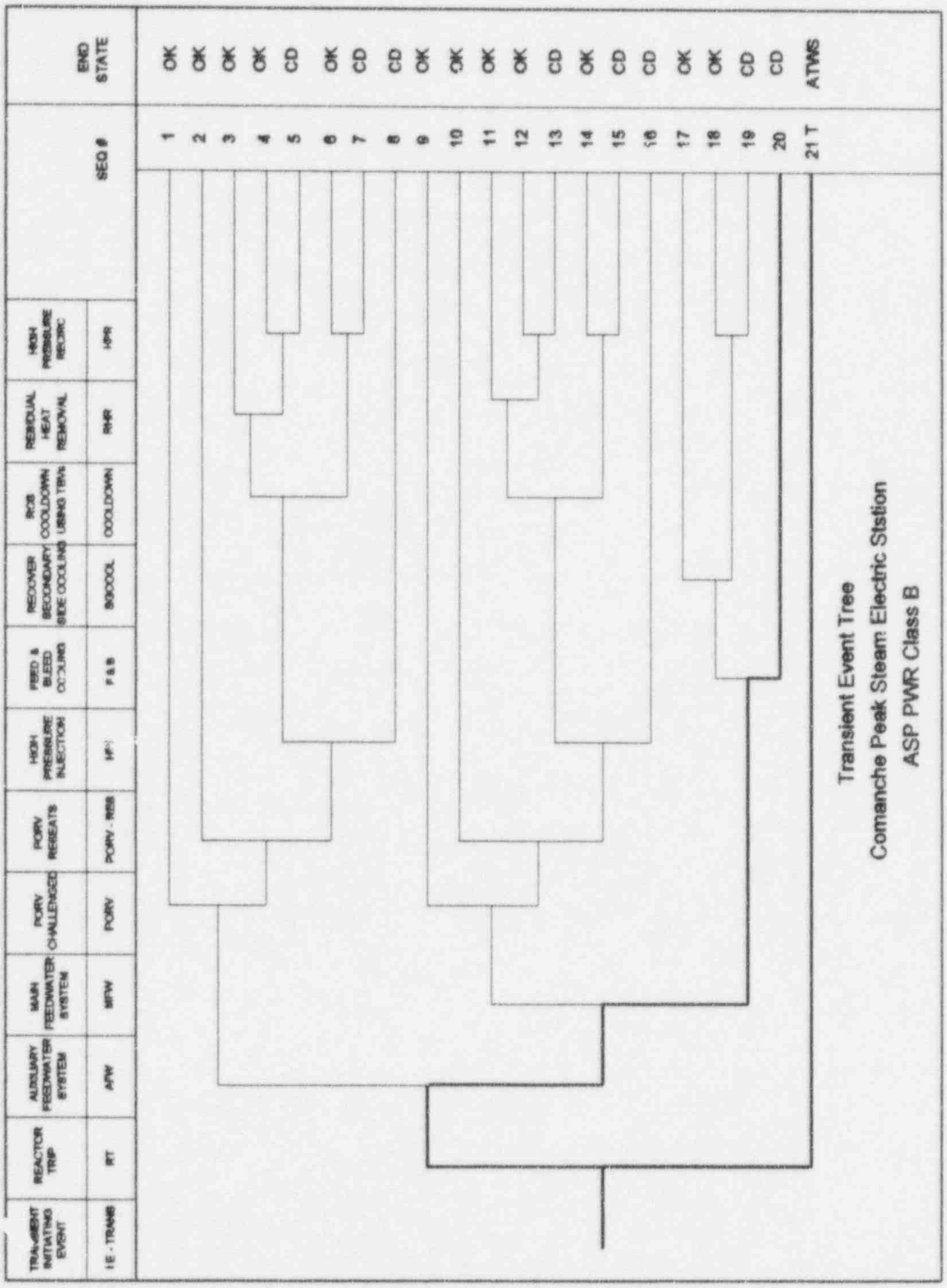

Fig. 3 Dominant core damage sequences for LERs 445/95-003, -004.

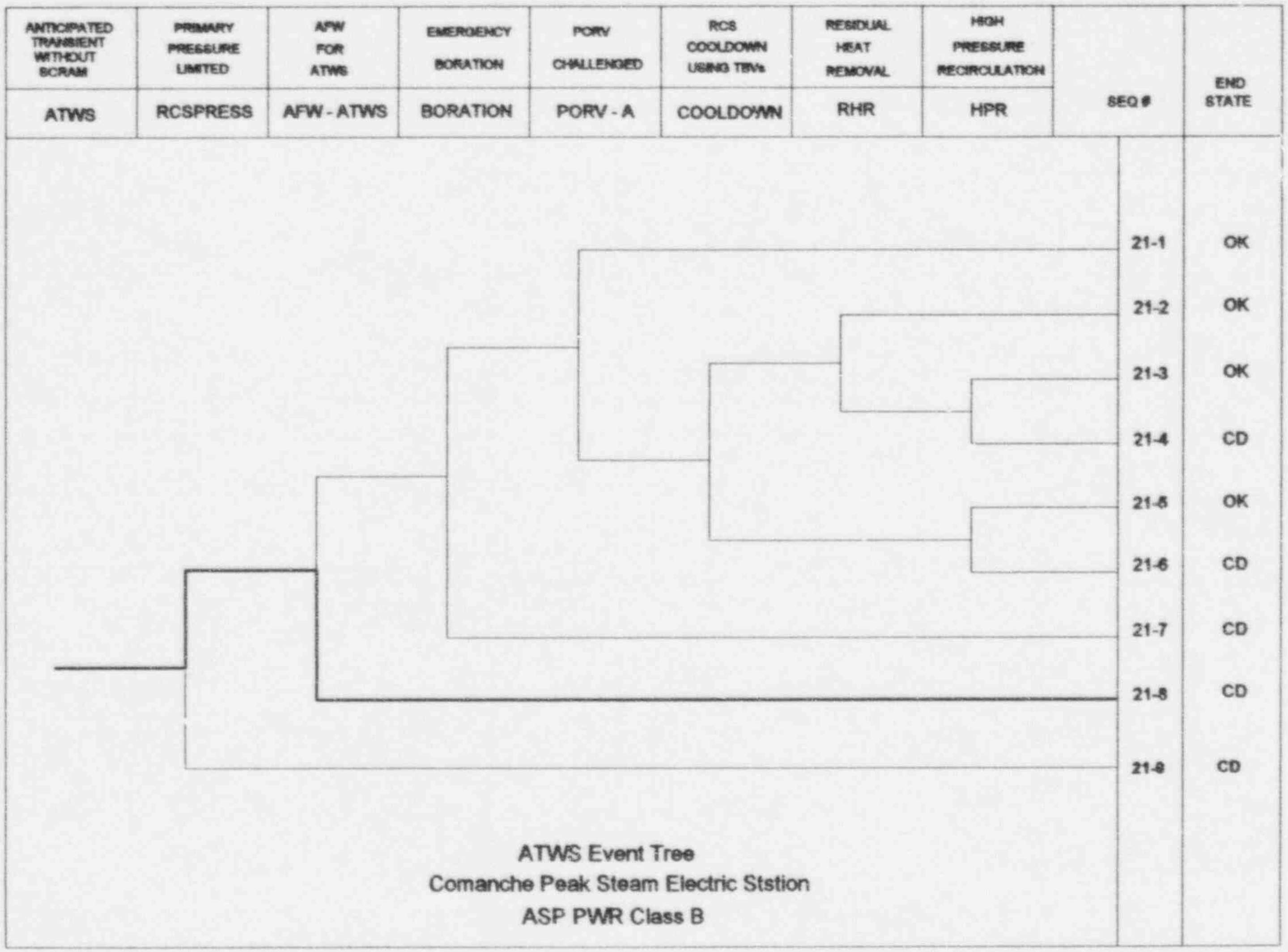

Fig. 4. Anticipated transient without scram (ATWS) event tree for Comanche Peak.

Table 1. Definitions and Probabilities for Selected Basic Events for LER 445/95-003, -004

Event name	Description	Base probability	Current probability	Type	Modified for this event
IE-LOOP	Loss of Offsite Power Initiating Event	8.5 E-006	0.0 E+000	IGNORE	No
IE-SGTR	Steam Generator Tube Rupture Initiating Event	1.6 E-006	0.0 E+000	IGNORE	No
IE-SLOCA	Small Loss of Coolant Accident Initiating Event	1.0 E-006	0.0 E+000	IGNORE	No
IE-TRANS	Transient Initiating Event	5.3 E-004	1.0 E+000		Yes
AFW-MDP -CF-AB	Common Cause Failure (CCF) of Motor-Driven Pumps	2.1 E-004	2.1 E-004		No
AFW-MDP-FC-AA	AFW Motor-Driven Pump A Fails During ATWS	4.0 E-003	1.0 E+000	TRUE	Yes
AFW-MDP-FC-1A	AFW Motor-Driven Pump A Fails	4.0 E-003	1.0 E-001		Yes
AFW-MDP-FC-1B	AFW Motor-Driven Pump B Fails	4.0 E-003	4.0 E-003		No
AFW-PMP-CF-ALL	AFW Serial Component Common to all Trains Fails (i.e., CCF)	2.8 E-004	2.8 E-004		No
AFW-TDP-FC-1C	AFW Turbine-Driven Pump Fails	3.2 E-002	1.0 E+000	TRUE	Yes
AFW-XHE-NOREC	Operator Fails to Recover AFW System	2.6 E-001	1.0 E+000	TRUE	Yes
AFW-XHE-NREC-ATW	Operator Fails to Recover AFW System During an ATWS	1.0 E+000	1.0 E+000		No
AFW-XHE-XA-SSW	Operator Fails to Align Suction to Service Water System (SSW)	1.0 E-003	1.0 E-003		No
HPI-XHE-XM-FB	Operator Fails to Initiate Feed and Bleed Cooling	1.0 E-002	1.0 E-002		No
MFW-SYS-TRIP	Main Feedwater System Trips	1.0 E+000	1.0 E+000		No
MFW-XHE-NOREC	Operator Fails to Recover Main Feedwater	2.6 E-001	1.0 E+000	TRUE	Yes

Table 1. Definitions and Probabilities for Selected Basic Events for LER 445/95-003, -004

Event name	Description	Base probability	Current probability	Type	Modified for this event
PPR-SRV-CC-1	Power-Operated Relied Valve (PORV) 1 Fails to Open on Demand	6.3 E-003	6.3 E-003		No
PPR-SRV-CC-2	PORV 2 Fails to Open on Demand	6.3 E-003	6.3 E-003		No
RPS-NONREC	Nonrecoverable RPS Trip Failures	2.0 E-005	2.0 E-005	NEW	Yes
RPS-REC	Recoverable RPS Failures	4.0 E-005	4.0 E-005	NEW	Yes
RPS-XHE-XM-SCRAM	Operator Fails to Manually Trip the Reactor	1.0 E-002	1.0 E-002	NEW	Yes

Table 2. Sequence Conditional Probabilities for LER 445/95-003, -004

Event tree name	Sequence name	Conditional core damage probability (CCDP)	Percent Contribution
TRANS	20	4.3 E-005	66.8
TRANS	21-8	2.0 E-005	31.2
Total (all sequences)		6.5 E-005	

Table 3. Sequence Logic for Dominant Sequences for LER 445/95-003, -004

Event tree name	Sequence name	Logic
TRANS	20	/RT, AFW, MFW, F&B
TRANS	21-8	RT, /RCSPRESS, AFW-ATWS

Table 4. System Names for LER 445/95-003, -004

System name	Logic
AFW	No or Insufficient AFW Flow
AFW-ATWS	No or Insufficient AFW Flow - ATWS
F&B	Failure to Provide Feed and Bleed Cooling
MFW	Failure of the Main Feedwater System
RCSPRESS	Failure to Limit RCS Pressure to <3200 psi
RT	Reactor Fails to Trip During Transient

Table 5. Conditional Cut Sets for Higher Probability Sequences for LER 445/95-003, -004

Cut set Number	Percent Contribution	Conditional Probability[a]	Cut sets[b]
TRANS Sequence 20		4.3 E-005	
1	22.9	1.0 E-005	AFW-XHE-XA-SSW, AFW-XHE-NOREC, MFW-SYS-TRIP, MFW-XHE-NOREC, HPI-XHE-XM-FB
2	14.4	6.3 E-006	AFW-XHE-XA-SSW, AFW-XHE-NOREC, MFW-SYS-TRIP, MFW-XHE-NOREC, PPR-SRV-CC-1
3	14.4	6.3 E-006	AFW-XHE-XA-SSW, AFW-XHE-NOREC, MFW-SYS-TRIP, MFW-XHE-NOREC, PPR-SRV-CC-2
4	9.2	4.0 E-006	AFW-MDP-FC-1A, AFW-MDP-FC-1B, AFW-TDP-FC-1C, AFW-XHE-NOREC, MFW-SYS-TRIP, MFW-XHE-NOREC, HPI-XHE-XM-FB
5	6.4	2.8 E-006	AFW-PMP-CF-ALL, AFW-XHE-NOREC, MFW-SYS-TRIP, MFW-XHE-NOREC, HPI-XHE-XM-FB
6	5.8	2.5 E-006	AFW-MDP-FC-1A, AFW-MDP-FC-1B, AFW-TDP-FC-1C, AFW-XHE-NOREC, MFW-SYS-TRIP, MFW-XHE-NOREC, PPR-SRV-CC-1
7	5.8	2.5 E-006	AFW-MDP-FC-1A, AFW-TDP-FC-1C, AFW-MDP-FC-1B, AFW-XHE-NOREC, MFW-SYS-TRIP, MFW-XHE-NOREC, PPR-SRV-CC-2
8	4.8	2.1 E-006	AFW-PMP-CF-AB, AFW-MDP-FC-1C, AFW-XHE-NOREC, MFW-SYS-TRIP, MFW-XHE-NOREC, HPI-XHE-XM-FB
9	4.0	1.7 E-006	AFW-PMP-CF-ALL, AFW-XHE-NOREC, MFW-SYS-TRIP, MFW-XHE-NOREC, PPR-SRV-CC-1
10	4.0	1.7 E-006	AFW-PMP-CF-ALL, AFW-XHE-NOREC, MFW-SYS-TRIP, MFW-XHE-NOREC, PPR-SRV-CC-2
11	3.0	1.3 E-006	AFW-MDP-CF-AB, AFW-TDP-FC-1C, AFW-XHE-NOREC, MFW-SYS-TRIP, MFW-XHE-NOREC, PPR-SRV-CC-1
12	3.0	1.3 E-006	AFW-MDP-CF-AB, AFW-TDP-FC-1C, AFW-XHE-NOREC, MFW-SYS-TRIP, MFW-XHE-NOREC, PPR-SRV-CC-2
TRANS Sequence 21-8		2.0 E-005	
1	98.0	2.0 E-005	RPS-NONREC, AFW-MDP-FC-AA, AFW-TDP-FC-1C, AFW-XHE-NREC-ATW
2	1.9	4.0 E-007	RPS-REC, RPS-XHE-XM-SCRAM, AFW-MDP-FC-AA, AFW-TDP-FC-1C, AFW-XHE-NREC-ATW
Total (all sequences)		**6.5 E-005**	

[a] The conditional probability for each cut set is determined by multiplying the probability of the initiating event by the probabilities of the basic events in that minimal cut set. The probability of the initiating events are given in Table 1 and begin with the designator "IE". The probabilities for the basic events are also given in Table 1.

[b] Basic events AFW-MDP-FC-AA, AFW-TDP-FC-1C, AFW-XHE-NOREC, and MFW-XHE-NOREC are all type TRUE events which are not normally included in the output of fault tree reduction programs. These events have been added to aid in understanding the sequences to potential core damage associated with the event.

LER No. 445/95-003, -004

Event Description:	Reactor trip, Auxiliary Feedwater (AFW) pump trip, second AFW pump unavailable
Date of Event:	June 11, 1995
Plant:	Comanche Peak 1

Licensee Comments

Reference: Letter from C. L. Terry, Texas Utilities Electric Company, to U.S. Nuclear Regulatory Commission, transmitting Comanche Peak Steam Electric Station (CPSES) - Unit 1 Docket Nos. 50-445 *Comments on Preliminary Accident Sequence Precursor Analysis of Reactor Trip at CPSES Unit 1 on June 11, 1995*, TTX-96397, July 15, 1996.

Comment 1: The quantitative values presented in this analysis are, in some cases, different from the values used in the CPSES Individual Plant Examination (IPE) study. In general, the core damage frequency contributions in the Preliminary Accident Sequence Precursor Analysis are based on somewhat conservative assumptions and more simplified models. Therefore, the numerical values should not be considered as accurate contributions to total core damage frequency. Rather, they should be used to determine the relative importance of various accident sequence precursors.

Response 1: Since its inception, this has been one of the primary objectives of the accident sequence precursor program. Over time, efforts have been made to make the models which are employed more realistic. However, it remains true that the ASP models are more simplified and potentially more conservative than those found in some IPEs.

Comment 2: The failure probabilities for the following events appear to be significantly higher than the values used in the CPSES IPE study.

- Operator fails to initiate Feed and Bleed cooling: a value of 1.0E-2 was used in this study versus 1.0E-3 used in the CPSES IPE.
- Failure of non-recoverable RPS trip: a value of 2.0E-5 was used in this study versus 1.0E-5 used in the CPSES IPE.

Response 2: Because of the sparseness of system failure events, data from many plants must be combined to estimate the failure probability of a multitrain system or the frequency of low- and moderate-frequency events (such as LOOPs and small-break LOCAs). Because of this, the

modeled response for each event will tend toward an average response for the plant class (refer to Table B.1 in NUREG/CR-4674, Vol. 21 for a listing of plants and their respective plant class). If systems at the plant at which the event occurred are better or worse than average (difficult to ascertain without extensive operating experience), the actual conditional probability for an event could be higher or lower than that calculated in the analysis. Regardless, the non-recoverable RPS trip value and the operator failure to initiate feed and bleed value are consistent with those values used in other probabilistic risk assessments (e.g., the Sequoyah PRA, the Farley IPE, the McGuire IPE) for all ASP plant classes for PWRs. Nevertheless, if the values for HPI-XHE-XM-FB (operator fails to initiate feed and bleed) and RPS-NONREC (nonrecoverable RPS trip failures) are changed to match the CPSES IPE values, the CCDP would be 3.7×10^{-5}. Even with these changes, the resulting CCDP is on the same order of magnitude and the event still meets the selection criteria as an ASP event (i.e., CCDP > 10^{-6}).

Comment 3: This event demonstrated that the CPSES operating crew was capable of recovering the Train A Motor-Driven Auxiliary Feedwater Pump (MDAFWP1-01) within 8 minutes and safely shutting down the plant. As a result, the failure probability to realign the Train A pump from a test configuration to the operating configuration should be very low. It should be noted that the Train A pump did not fail to operate and, therefore, the recovery here did not require repairing a failed pump but rather realigning Train A to an operating configuration. Consequently, the probability of not successfully realigning the pump under the given conditions should be between 1E-2 and 1E-4.

This recovery, on the other hand, should have a failure probability of 1.0 for an Anticipated Transient Without Scram (ATWS) event since the pump is required to be available almost immediately.

Response 3: The non-recovery value was estimated using the methodology described by Sattison (Methods Improvements Incorporated into the SAPHIRE ASP Models, Sattison et. al., NUREG/CP-0140, Vol. 1, Proceedings of the Twenty-Second Water Reactor Safety Information Meeting, October 1994, USNRC).

In response to this comment, a more rigorous approach was taken. Non-recovery as a function of time was modeled as being lognormally distributed (see E. M. Dougherty and J. R. Fragola, *Human Reliability Analysis*, Wiley and Sons, 1988), with a median response time of 8 minutes (actual response time) and a window of 30 minutes. Assuming a burdened-recovery error factor of 6.4, the probability of non-recovery within 30 minutes is approximately 0.1, the same as estimated by the Sattison approach.

Since it was assumed that the entire 30 minutes would be dedicated to recovery of AFW-MDP-FC-1A, the system nonrecovery, AFW-XHE-NOREC, was set to 1.0.

As described in the modeling assumptions section, motor-driven AFW pump A was assumed to be inoperable for ATWS events.

Comment 4: The cut set numbers 1, 2, and 3 of TRANS Sequence 20 in Table 5 have the basic event AFW-XHE-XA-SSW for which there is no definition. This basic event appears to be the failure of the operator to align the suction of the AFW pumps to the Station Service Water (SSW) system. This capability exists at CPSES Units 1 and 2. However, it is normally in a locked closed position and it is only required when the normal water source of the Condensate Storage Tank (CST) is not available. Since the failure probability of the CST is very low, the alternate SSW water source was not credited in the CPSES IPE study. Nevertheless, if it is modeled, the corresponding cut sets should also include the failure of CST. In cut sets 1, 2, and 3, the CST term is not included, but SSW alignment is included.

Response 4: Basic event AFW-XHE-XA-SSW is defined in Table 1 as "Operator Fails to Align Suction to the Service Water System (SSW)." However, this event would be better described as "Operator fails to align makeup to the CST."

Comanche Peak's condensate storage tank (CST) has a volume of approximately 500,000 gal. Technical Specifications (3.7.1.3) require a minimum tank level of 53% be maintained. Assuming that tank level correlates linearly with volume, this corresponds to about 265,000 gal. The FSAR indicates that 282,000 gal will be dedicated for AFW operation and the IPE makes a similar assumption. This reserved capacity of the CST is good for 9 h—maintaining hot standby for 4h and then to cooling down to conditions which would permit alignment to the RHR.

The ASP model for AFW success requires sufficient inventory for up to 24 h of operation. A simple calculation was performed using the Untermyer-Weills decay heat correlation to estimate the amount of CST inventory that would be required to remove this decay heat without replenishment (Reference: S. Glasstone and A. Sesonske, *Nuclear Reactor Engineering*, D. van Nostrand, 1967). It was estimated that about 2.2E+9 BTUs would be rejected, requiring about 330,000 gal of CST inventory. This quantity is significantly more than the amount required by technical specifications or assumed in the IPE.

The actual level in the CST will fluctuate from near maximum (500,000 gal) to near the technical specification limit. Without further knowledge of Comanche Peak's operating practices, it is impossible to determine when sufficient inventory exists in the CST and when it does not. Regardless, any reasonable assumption will not greatly affect the CCDP calculated for this event. For example, if it is assumed that the CST inventory would be inadequate 50% of the time, those cut sets containing the basic event AFW-XHE-XA-SSW would be weighted by 0.5 (specifically, cut set numbers 1, 2, and 3 in TRANS Sequence 20). This change would reduce the estimated CCDP from 6.5E-5 to 5.2E-5. Hence, the event still qualifies as an ASP type event.