September 1, 1981

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

IN the Matter of

TEXAS UTILITIES GENERATING COMPANY, ET AL.

(Comanche Peak Steam Electric Station, Units 1 and 2)

Docket Nos. 50-445
50-446

(Application for Operating License)

CFUR'S FIRST AND SECOND SET OF SUPPLEMENTAL ANSWERS TO NRC STAFF'S FIRST AND SECOND SET OF INTERROGATORIES

Citizens for Fair Utility Regulation (CFUR) files this its partial supplemental answers to NRC Staff's 1st and 2nd set of Interrogatories, pursuant to Board Order.

CFUR files answers for the following interrogatories:

C1-3, Second supplemental answer

C3-2, Second supplemental answer

C3-3, Second supplemental answer

C3-4, Second supplemental answer

C3-5, Second supplemental answer

C3-10, First supplemental answer

C3-11, Second supplemental answer

C3-12, First supplemental answer

C3-19 through 22, First supplemental answers

C4-10, First supplemental answer

C4-12, First supplemental answer

C4-15, First supplemental answer

C9-3, 4 and 6, second supplemental answers

CFUR will file additional answers to additional interrogatories forthwith.

D503
S0/1

2nd Supplementary Answer

A) "Agreement of Settlement ('Agreement') dated December 16, 1977 between Westinghouse Electric Corporation and Owners of Comanche Peak.
From page 13 of Agreement: "Westinghouse will provide to Owners free of charge the effort required for the preparation and defense of the Comanche Peak FSAR sections within the Westinghouse scope of responsibility in accordance with the requirements of Regulatory Guide 1.70, Revision 2, 'Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants - LWR Edition', dated October 16, 1975. ('Rev. 2')"
From page H-6 of Agreement: "Vendor, free of charge to Purchaser, shall implement the provisions of ... Rev. 2 for those portions of the FSAR for Comanche Peak which are within Vendor's scope of supply as identified in Revision 6 of the FSAR Scope Split dated November 23, 1977. ...
Vendor's portion of the Rev. 2 changes are in Chapters 1, 3, 4, 5, 6, 7, 9, 11, 12, 15, and 16 of the FSAR."

B) CFUR asserts that the Applicants relied on Westinghouse to originate the following portions of the CPSES/FSAR:
Sections 1.5, 1.6, Appendix 1AN, 3.6N.1, 3.6N.2, 3.7N, 3.9N.1, 3.9N.2, 3.9N.3, 3.9N.4, 3.9N.5, 3.10N, 3.11N, 4.1, 4.2, 4.3, 4.4, 4.5, 4.6, Appendix 4A, 5.1, 5.2.1, 5.2.2, 5.2.3, 5.2.4, 5.3, 5.4.1, 5.4.2.1, 5.4.2.2, 5.4.3, 5.4.4, 5.4.6, 5.4.7, 5.4.10, 5.4.11, 5.4.12, 5.4.13, 5.4.14, 6.1N, 6.3, 7.1, 7.2, 7.3, 7.4, 7.5, 7.6, 7.7, 9.1.1, 9.1.2, 9.1.4, 9.3.4, 11.1, 11.3, 15.1.1, 15.1.2, 15.1.3, 15.1.4, 15.1.5, 15.2.1, 15.2.2, 15.2.3, 15.2.4, 15.2.5, 15.2.6, 15.2.7, 15.2.8, 15.3.1, 15.3.3, 15.3.4, 15.4.1, 15.4.2, 15.4.3, 15.4.4, 15.4.6, 15.4.7, 15.4.8, 15.5.1, 15.5.2, 15.6.1, 15.6.2, 15.6.3, 15.6.5, 15.7.1, 15.7.2, 15.7.3, 15.7.4, 15.8, 15.8.1, 15.8.2, 15.8.3, 15.8.4, 15.8.5, 15.8.6, 15.8.7, Appendix 15A.

C) CFUR asserts that the Applicants relied on Westinghouse to provide input, review and comment on the following portions of the CPSES/FSAR:
Sections 1.2.2, 1.3.1, 1.3.2, 1.4, 1.7, 3.1, 3.2. 3.5, 3.8, 5.2.5, 6.2.1, 6.2.5, 8.3.1, 9.4, 11.2, 11.4, 12.1.2, 12.2.1, 12.2.2, 12.4, 16.2, Appendix 17A

D) CFUR asserts that the Applicants relied on Westinghouse to review the following portions of the CPSES/FSAR:
Sections 1.1, Appendix 1AB, 3.7B.1, 3.7B.2, 3.7B.3, 3.7B.4, 3.7BA, 3.9N.6, 3.9B.1, 3.9B.2, 3.9B.3, 3.9B.4, 3.9B.5, 3.9B.6, 3.10B, 3.11B, 5.4.5, 5.4.9, 6.1 , 6.2.2, 6.2.3, 6.2.4, 6.4, 6.5, 8.1, 8.2, 8.3.2, 8.3.3, 9.1.3, 9.2.5, 9.3.3, 9.5.1, 9.5.2, 9.5.3, 9.5.4, 9.5.5, 9.5.6, 9.5.7, 9.5.8, 10.3, 10.4.1,

C1-3 (cont)

10.4.2, 10.4.3, 10.4.4, 10.4.5, 10.4.6, 10.4.7, 10.4.8, 10.4.9, 10.4.10, 10.4.11, 10.4.12, 10.4.13, 10.4.15, 10.4.16, Appendix 11A, 12.3.1, 12.3.2, 12.3.3, 12.3.4, 14.2.1, 14.2.2, 14.2.3, 14.24, 14.2.5, 14.2.6, 14.2.7, 14.2.8, 14.2.9, 14.2.10, .4.2.11, 14.2.12, 15.7.5, 17.1, 17.2, 10.4.14.

C3-2

In addition to CFUR's first supplemental answer, CFUR responds:

In NUREG-0737, "Clarification of TMI Action Plan Requirements," Task I.C.1, the NRC staff found what they describe as "recurring deficiencies" in the guidelines being developed for the evaluation and development of procedures for transients and accidents. The staff comments:

"Specifically, the staff has found a lack of justification for the approach used in developing diagnostic guidance for the operator and in procedural development. It has also been found that although the guidelines take implicit credit for operation of many systems or components, they do not address the availability of these systems under expected plant conditions nor do they address corrective or alternative actions that should be performed to mitigate the event should these systems or components fail...." (P. I.C.1-2).

"The analyses conducted to date for guideline and procedure development contain insufficient information to assess the extent to which multiple failures are considered. NUREG-0578 concluded that the single-failure criterion was not considered appropriate for guideline development and called for the consideration of multiple failures and operator errors. Therefore, the analyses that support guideline and procedure development should consider the occurrences of multiple and consequential failures. In general, the sequence of events for the transients failures such that, if the failures were unmitigated, conditions of inadequate core cooling would result.

"Examples of multiple failure events include: Multiple tube ruptures in a single steam generator, and tube rupture in more than one steam generator; failure of main and auxiliary feedwater; failure of high-pressure reactor coolant makeup system; an anticipated transient without scram (ATWS) event following a loss of offsite power, stuck open relief valve or safety/relief valve, or loss of main feedwater; and operator errors of ommission or commission.

"The analyses should be carried out far enough into the event to assure that all relevant thermal/hydraulic/neutronic phenomena are identified...Failures and operator errors during the long-term cooldown period should also be addressed.

"The analyses should support development of guidelines that define a logical transition from the emergency procedures into the inadequate core cooling procedure including the use of instrumentation to identify inadequate core cooling conditions. Rationale of this transition should be discussed. Aditional information...includes:

1.A detailed description of the methodology used...
2.Associated control function diagrams, sequence-of-event diagrams, or others, if used;
3....bases for mulitple and consequential failure considerations;
4.Supporting analysis, including a description of any computer codes used; and
5....description(s) of the applicablity of any generic results to

C3-^ cont.

plant specific applications....

"Pending staff approval of the revised analysis and guidelines, the staff will continue the pilot monitoring of emergency procedures described in Task Action Plan item I.C.8 (NUREG-0660.) For PWR's this will involve review of the loss of coolant steam-generator-tube rupture, loss of main feedwater, and inadequate core cooling procedures."

The modified computer codes must be tested to obtain a measure of absolute accuracy of the calculated values with respect to some recognized standard such as the small-break LOCA sequences tested with Semiscale and/or LOFT. The margins of safety (e.g., if operator had waited an additional length of time, the allowable cladding temperature would be exceeded) [redacted] are required to be determined.

From 10 CFR 50.34 (a)(4) and (b)(4): "A final analysis...with the objective of assessing the risk to public health and safety resulting from operation of the facility and including determination of (i) the margins of safety during normal operations and transient conditions anticipated during the life of the facility and (ii) the adequacy of structures systems, and components provided for the pervention of accidents...and taking into account any pertinent information developed since the submittal of the preliminary safety analysis report."

From 10 CFR 50.34 (b) and (b)(2): "The final safety analysis report shall include information that presents a safety analysis...of the facility as a whole, and shall include... evaluation...to show that safety functions will be accomplished."

Failure to modify computer codes to account for the parameters cited in NUREG-0737 will lead to adoption of codes that will not realistically predict plant behavior. In addition, a number of other tasks in NUREG-0737 and the Applicant's compliance to them in NUREG-0797 are relevent to this contention:

Part A

I.A.1.1. Shift Technical advisor - potential for operator error
I.B.1.2. Independent Safety Engineering Group -- operator error.
I.A.2.1. Immediate upgrading of training - operator error.
II.B.1. Reactor Coolant System vents - noncondensible gases
II.B.2. Design Review Systems used in Postaccident Operations - Hydrogen and noncondensible gas levels.
I.D.1 Control room design features - operator error, misleadint indications.
II.D.3 Direct Indication of Relief and Safety Valve Positions - Misleading indications.
II.E.1.1 Auxilliary feedwater System Evaluatin - maintenence errors, misleading indications
II.E.3.1. Emergency Power - Pressurizer Heaters - Maintenence errors

C3-2, cont.

II.F.1. Accident Monitoring Instrumentation - operator error plus misleading indications
II.F.2. Instrumentation, Inadequate Core Cooling - operator error and misleading indications
II.K.1.10 Procedures, Operability status of maintenance items - operator and/or maintenance errors

SMALL BREAK

II.K.2.13 LOCA W/O Auxilliary Feedwater--analysis of Small Break LOCA's
II.K.2.6 Reactor Coolant Pump Seal Damage -- limits validity of small break LOCA model
II.K.2.17 Potential for voiding in RCS during Transients - Small Break LOCA
II.K.3.1 Installation of Automatic PORV Isolation System - small break LOCA
II.K.3.2 Overall Safety Effect of PORV Isolation System - small break LOCA
II.K.3.5 Automatic Trip of RCP during LOCA - maintenance error small break LOCA
II.K.3.30 -Revised smallbreak LOCA methods to show compliance with 10 CFR 50 Appendix K
IIK. 3.31. Plant Specific Calculations for CPSES -small break LOCA's
II.K.3.46 Natural Circulation in Depressurization of RPU during small break LOCA's.

C3-3

The computer codes utilized in the accident sequence analyses contained in the CPSES/FSAR, Chapter 15, do not in general, sufficiently account for hydrogen generation in the primary coolant loop, operator errors, maintenance errors, multiple failures, consequential failure, equipment failures of a secondary nature, misleading indications from instrumentation and central systems, non-condensable gases and small break LOCA's. As a result, these analyses, in general, are deficient to the point of providing inaccurate answers.

C3-4

This response supersedes CFUR's supplemental response to interrogatory C3-4:

The computer codes addressed in Contention 3 are the following: LOFTRAN, SATAN, LEOPARD, FACTRAN, COCO, TACT, THINC, TWINKLE, LOCTA, WREFLOOD, TURTLE, WFLASH.

CFUR contends that the following computer codes used in Section 15 of the CPSES/FSAR, in general, produce inaccurate answers for the following reasons:
1.) There are insufficient allowances for operator and/or maintennance error.
2.) The single-failure criteria interpretation used in these codes are too restrictive in that they do not sufficiently allow for analysis of multiple failures, consequential failure, and equipment failures of a secondary nature.
3. The codes do not sufficiently predict the consequences of a small break LOCA.
4.) There are insufficient allowances for the formation of hydrogen and uncondensible gases in the primary coolant loop.
5.) The computer codes do not sufficiently allow for the ramifications form the intrumentation and central system, particularly with regard to operator error.

C3-5

See NUREG-0737, item I.C.1. Also see response to interrogatory C3-2, part A. These items comprise "pertinent information" under 10 CFR 50.34 (b) (4). Failure to include them in analysis of accident sequences will lead to use of computer codes providing inaccurate answers, as they pertain to parameters relevant to operations at CPSES.

C3-10

List of relevant parameters including but not limited to:
1. Operator errors. Errors of ommission and commission based on insufficient or inaccurate information and/or failure to properly interpret data to mitigate accident and transient consequences at the proper point.

C3-10, cont.

2.) Maintenance errors. Failure to maintain equipment which could impact on safety functions in the primary coolant loop, the secondary loop and the engineered safety features.
3.) Hydrogen formation. Creation of hydrogen in the primary coolant loop.
4.) Single Failure criteria-interpretation. Two or more independantly occuring single failures that, if left unmitigated, would result in conditions of inadequate core cooling.
 (b.) Consequential failure--those failures that occur as a result of one or more single failures.
 (c.) Equipment failures of a secondary nature--failures of equipment to perform unspecified yet assumed functions in an accident sequence.
5.) Misleading indications. Failures of reactor instrumentaion systems to provide accurate and reliable operational data as well as the absence of direct indications.
6.) Noncondensable gases. Creation and/or release of noncondensable gases that are contained in the primary coolant loop.
7.) SMall-break LOCA consequences. Enclosure 1, pages 9 through 12 of "Report of CFUR's Position on Each Contention", April 10, 1980, documents CFUR's basis as to the following:

> "The frequency of the TMI accident is much greater than one in a million and would be categorized as a credible accident of either the infrequent variety or the limiting fault variety. The two accident sequences analyzed in the CPSES/FSAR which most closely rewembles the start of theTMI sequence are the loss of normal feedwater flow sequence and/or the feedwater system pipe break sequence.
> Those sequences utilize the LOFTRAN, FACTRAN, and THINC computer codes. Yet,none of these codes contain the capability of determining the amount of hydrogen generated during the accident sequence...."

8.) TMI-2 accident description. The accident involved three different elements:
 (a.) Maintenance Error: A number of maintenance activities were intertwined with the TMI-2 accident. The actions of the foreman and two auxillery operators in an attempt to unclog a resin plug in a pipe leading from a condensate polisher iniitated the sequence of events. Prior maintenance activities also were contributing factors...the feedwater valves which were improperly left closed are obvious contributors...In addition... sluggish response on the part of maintenance to cure problems with instrumentation in the control room also contributed, especially when it is recognized that maintenace tags obscured the view of some of the instruments which were operating normally.
 (b.) Equipment Failure: Although the pilot-operated relief valve performed normally when it opened, it failed to close properly which will be referred to as its secondary function.
 (c.) Operator Error: Besides not being trained for this accident sequence, the material available to operators had apparently convinced them that the sequence of events which occured

Q3-10, cont.

at TMI-2 were improbable. The number of errors are lengthy and complex, and would take long to enumerate...certain contributors to these errors was the fact that the FSAR was deficient but never challenged:

"Based on our training, it was impossible...if you look back through everybody's training and theFSAR and safety analysis and thebuilding construction, you will not see a paragraph that projects thattype of transient. (It) is so particularly foreign and unblievable that it has absolutely no significance. That's why nobody did anything about for two days. (NRC Special Inquiry Group, Three Mile Island-A Report to the Commissioners and to the Public, Mitchell Rogovin, Director, January 1980, p. 43.)

CFUR's Partial Substantive Objections to Applicants' Statement of Objections, July 23, 1980, further documents that NUREG-0578 mandates that "further analyses of small LOCA's are needed" and that "more and a different kind of analysis of accident analyses is needed." According to "Clarification of TMI Action Plan Requirements", NUREG-0737, Task I.C.1, p.IC.1-2, "NUREG-o578 concluded that the single-failure criteria was not considered appropriate for guideline development (guidance for the evaluation and development of procedures for transcients and accidents) and called for the consideration of multiple failures and operator errors. Therefore the analyses that support guideline and procedure development should consider the occurences of multiple and consequential failures. In general, the sequence of events for the transcients and accidents and inadequate core cooling analysis should postulate multiple failures such that if the failures were unmitigated, ...inadequate core cooling would result...the analyses should be carried out far enough into the event to assure that all relevant thermal/hydraulic/Neutronic phenomena are identified...failures and operator errors during the long term cooldown preiod should also be addressed. The analyses should support development of guidelines that define a logical transition from the emergency procedures into the inadequate core cooling procedure including the use of instrumentation to identify inadequate core cooling conditions."

It should be note that in "CFUR's 4th Set of Interrogatories to Applicants" that in answer to interrogator 12, the Applicants do <u>not</u> take exception to items I.C.1., II.K.3.30, and II.K.3.5.

According to 10 CFR 50.34 (b) (4), " A final analysis and evaluation of the design and performance of structures systems and components with the objective stated in paragraph (a) (4) [regarding assesement of risk to public health and safety and adequacy of structures, etc., provided for the prevention of accidents and the mitigation of accident consequences] of this section and taking into account any pertinent information developed since the submittal of the preliminary safety analysis report."

In order to comply with 10CFR 50.34 (a)(4) and(b)(4), it will be necessary to perform the analysis with "pertinent information"

C3-10. cont.

regarding the parameters reflecting the sequence of events at TMI.

C3-11

Operators errors will lead to inadequate responses to transient and accident conditions due to failure to properly interpret data or to read inadequate data which may be misleading. Maintenance errors will lead to failures in systems and components which could impact on safety funcitons, inhibiting or eliminating their response to transcient and/or accident conditions.
Hydrogen generation would lead to potentially explosive mixtures which could wreck containment components or could inhibit core cooling by blockage of coolant intakes or pump cavitation.
Single failure criteria interpretation in too narrow a sense does not prevent development of multiple failure and consequential failure hazards which could disable or seriously restrain systems and components that impact on safety functions. Equipment failures of a secondary nature may undermine the consistency and effectiveness of primary function responses.
Misleading indications could induce operator error or could affect automatic operations of systems impacting on safety functions.
Noncondensable gases could lead to coolant flow blockage, pump cavitations and creation of flux tilt conditions which may lead to partial core melt.

According to NUREG-0737, Task I.C.1, p.I.C.1-2, NUREG-0578 concluded that the single-failure criterion did not appropriately account for multiple failures and operator errors, nor did it account for consequential failure As a result, new sequences of events should postulate multiple failures, consequential failures and operator errors such that failure to mitigate would create conditions of inadequate core cooling. Accident sequences should be carried out' sufficiently far enough to identify relevant phenomena...and should consider use of instrumentation to identify inadequate core cooling conditions.
Analyses of small break LOCA's are required under TASK II.K.3.30, p. II.K.3.30-1 of NUREG-0737. Both I.C.1 and II.K.3.30 are required of applicants for operating license, including those for CPSES. The Applicants have acknowledged no exceptions to those requirements in their answers to CFUR's 4th set of interrogatories.

C3-12

"Realistically predict plant behavior" means forecast the action or reaction of CPSES with a high level of confidence. The accident sequence analyses supplied for the Applicant at the time this contention was written were clearly inadequate to provide a proper basis for plant design and for the development of operator training programs and operating procedures. The applicant failed to analyse more than the initial minutes of a transient, wheras such analyses should have covered a time period until a stable system had been assured.

C3-12, cont

The Applicant has failed to take into account maintenance errors as they affect both safety-grade and non-safety grade materials . The applicant has failed to account for operator errors that compound errors. The Applicant has failed to consider multiple failures and consequential failures in their accident analyses. In Addtion, the applicant has failed to account for problems with non-condensible gases in the primary coolant loops. The Applicant has also failed to compensate for misleading indications for the instrumentation and c system.

CFUR contends that those changes proscribed in NUREG-0737, items I.C.1, II.K.3.30 and II.K.3.5 constitute "pertinent information" under 10 CFR 50.34 (b) (4). The Nuclear Regulatory Commission has proposed for licensing requirements rules on May 13, 1981, which incorporate item I.C.1 recommendations for "Analyses of small-break LOCAs and of transients and accidents that involve postulated multiple failures, consequential failures and operator errors which if unmitigated, could lead to inadequate core cooling". CFUR' contends the new NRC requirements are an attempt to recognize these analyses as "pertinent information" within the scope of 10 CFR 50.34(b)(4).

This proposed rule also acknowledges the "pertinence" of carrying an analysis suffidently into the event to "identify all significant phenomena" and "address possible failures and operator errors during the long-term cooling phase." Also, by proscribing analyses that "support development of guidelines that define a logical transition from the emergency procedures into the inadequate core cooling precedures including the use of instrumentation to identify inadequate core cooling conditions." If failures of instrumentation systems lead to creation and usage of misleading indications of core and primary coolant loop conditions, analysis of accident sequences with inadequate core cooling conditions are undermined sufficiently to prohibit realistic response to transient conditions. Consideration of the effects of misleading indications is crucial to the development of effective and "pertinent" accident analyses. Moreover, development of analyses under item I.C.1, would be inhibited by a failure to account for hydrogen formation in the primary coolant loop. Since such formations were shown to have blocked coolant flow at TMI, procedures to identify inadequate core cooling conditions and to mitigate their consequences would be deficient and would fail to meet the criteria of 50.34 (b) (4) in providing a sufficient analysis of "structures, systems, and components provided for the prevention of accidents and the mitigation of the consequences of accidents."

Development of accident analyses under item I.C.1, would be inhibited by a failure to account for noncondensible gases in the primary coolant loop. Since such formations were shown to have blocked coolant flow at TMI, procedures to identify inadequate core cooling conditions and to mitigate their consequences would be deficient and would fail to meet the criteria of 50.34 (b) (4).

Maintenance errors are "pertinent information" as they create potential for "postulated multiple failure, consequential failures, and operator errors, which if unmitigated, could lead to

C3-12, cont/

inadequate core cooling." On May 12, 1981, in a letter to the NRC, the advisory committee on Reactor Safeguards argued that "a supporting infrastructure of procedures, information and trained personnel is as important to safe operation of nuclear power plants as capable licensed operators." The ACRS said, it "has become increasingly aware of a lack of requirements for such support systems." Areas outlined by ACRS as its key concerns are: Availability of knowledgeable maintenance personnel who are familiar with plant-specific hardware...qualifications, such as training for service personnel responsible for instrumentation, electrical distribution, fluid system testing computer software, and water treatment functions; criteria for decisions, or the need for repair modification or replacement of equipment with observed in-service deficiencies; planning for procedural and record-keeping practices for maintenance and other service activities, work, controls and communications practices for service activities that may adversely affect public safety as a reslut of errors; completeness and accessibility of maintenance and servicing information for power plant equipment; and conditions for using contract maintenance and service personnel in place of trained licensee employees for supporting service functions.

Although CFUR does not consider the ACRS problem areas as wholly inclusive, we do consider these areas to be "pertinent information" under 10 CFR 50.34 (b) (4). CFUR contends that the use of "pertinent information" as defined in our basis, will lead to development of accident analyses and utilized computer codes that have the potential to "realistically predict plant behavior."

In Addition, CPSES is the first Westinghouse plant to attempt to operate without a boron injection tank. On a normal Westinghouse PWR, the highest head source of water available is provided by the CVCS charging pumps and therefore, it is the most responsive safety injection system during high-pressure transients or accidents.If the boran injection tank is deleted, the planned safety margin would be compromised.

The Applicants position ("Summary of Meeting on CP Design Change and Responses to RSB Questions" by S.B. Burwell of the NRC Staff, May 26,1981) is that "the steam line break is the only design basis accident (Chapter 15) for which credit is taken for the boran injection tank." CFUR contends that the true value of a high break system is during a high pressure accident which may arise as the result of a small break or partial fuel blockage. Example: While the boran injection tank may not be taken credit for in any accident sequence other than the large-break LOCA, the operator is left without the margin of safety provided by the high-head safety injection system on occasions when non-automatic actions are called for during a high-pressure accident.

C3-19

Yes.

C3-20.

We intend to challenge the accuracy of the following codes in the FSAR:

LOFTRAN, COCO, TACT, FACTRAN, TWINKLE, LOCTA, THINC, TURTLE, SATAN, WREFLOOD, LEOPARD, WFLASH.

C3-21

In general, these computer codes are not able to accept the following parameters:

1.) Operator errors: No evidence of allowance for errors of ommission or commission.
2.) Maintenance errors: No evidence of allowance for errors in maintenance of primary, secondary and engineered safety features.
3.) Single failure criteria: Too strict an interpretation applied to the criteria leading to exclusion of multiple failures, consequential failures and equipment failures of a secondary nature.
4.) Small break LOCA:Inability to predict consequences accurately.
5.) Hydrogen formation: Inability to analyze hydrogen generation in terms of adequacy of core cooling.
6.) Noncondensible gases: No evidence of allowance for noncondensible gases in the primary coolant loop.
7.) Misleading indications: Inability to analyze effects of misleading indicators from instrumentation systems.

C3-22

The Applicants have not presented sufficient or satisfactory evidence that any accident sequence in Section 15 of the FSAR has adequately compensated for these parameters. In fact, in the case of many of the stated parameters there is no clear evidence that the applicant has addressed the parameters in any substantive fashion. The Applicants have consistently chosen a representative accident to analyze, ignoring accidents they consider less credible or probable. They have used a narrow set of parameters to be examined.Mostly those involved physical conditions within the core, primary coolant loop, pressurizer, etc. Yet, there is no clear evidence of a systematic attempt to incorporate implications of operator or maintenance error or the operational status of structures, systems and components designed to maintain the proscribed conditions. By restricting their analyses to a simple single failure transient or accident, they fail to concurrent failures, consequential failures or failures of equipment to return to pre-event status, thus threatening the credibility of an analysis based on narrow sets of parameters and credible failures.

The Applicants have not provided an analysis of a small break LOCA that embodies enough of the parameters to permit realistic predictions of CPSES behavior.

C3-22, cont.

NUREG-0737 includes a series of items (listed in response to Interrogatory C3-2) that address issues involving our parameters. Since these items arose after TMI-2, it is not likely that they were addressed sufficiently in general, to permit incorporation of these items in pre-TMI-2 computer codes and corresponding accident sequences. Comparing these items with the Applicants responses to them in NUREG-0797, the Safety Evaluation Report for CPSES, a number of points are made.

First, critical assumptions regarding conditions and operational status of structures, systems and components at CPSES were made. If those assumptions are undermined by the items of NUREG-0737, the potential for existence and significence of our parameters is increased.

Second, many of the items in NUREG-0737 have been dealt with in an insufficient manner, as recorded in NUREG-0797. Among those in dispute are I.A.1.1, I.B.1.2, I.C.I., II.B.2, and II.F.1. A complete list is found in response to C3-2, part A.

Third, the Applicants have not presented sufficient evidence to show that the computer codes utilized in the CPSES/FSAR account for the items listed from NUREG-0737 or correspondingly for the parameters previously stated that may arise and show significnat results of the failure to compensate for those in Part A.

CFUR contends that failures to provide timely resolution of the problems posed by the items of Part A leads to the creation of the stated parameters, although they are not the only source of deviation. At minimum, the computer codes must be adjusted to provide for a comprehensive analysis of Part A items and their corresponding parameters to permit "realistic" predictions of plant behavior.

C9-3

Interrogatory Do you assert in Contention 9 that as a result of operation of Comanche Peak, there will be "radioactive releases"? If so, state the basis for such assertion. Do you contend that such "radioactive releases" will result from normal operation or only as a result of an accident? State the basis for your position in this regard.

Original Answer a) yes. FSAR Sections 12 and 15, Contention 4 basis and the series of NUREG-0521 reports, "Radioactive Releases from Nuclear Power Plants (Year)".

b) Yes. Same as (a)

Motion To Compel Although the first part of CFUR's answer to this Interrogatory contains some of the information which the Staff seeks in Interrogatory C9-3, CFUR's response does not answer the second question posed in that interrogatory. By merely answering "Yes" to that question, CFUR does not state, as is requested, whether the radioactive releases with which it is concerned will result from normal operation of CPSES or only as a result of an accident. Accordingly, CFUR should be directed to answer that question.

Supplementary Answer (5/22/81) Some radioactive releases will result from normal operation, and it cannot be ruled out that some will result from transients, accidents or incidents.

Telephone Contact 8/18/81 NRC Staff acknowledges that Interrogatory has been properly supplemented. Staff refuses additional oral voluntary disclosure.

Voluntary Written Disclosure The object of this contention is to insure that planned batch releases of radioactive gases will be accomplished during meteorological conditions which minimize radiation exposures. During normal operation and anticipated operational occurrences, some radioactive gas will be accumulated which will, on occassion, be released in a planned manner to the atmosphere. During some transient, accident and/or incident sequences a somewhat larger amount of radioactive gas may be successfully accumulated (not released to the atmosphere). It is possible that some of these radioactive gases will also be released to the atmosphere in a planned manner.

10CFR§20.1(c) states that the applicant should, in addition to complying with the requirements set forth, make every reasonable effort to maintain radiation exposures as low as reasonably achievable.
Action taken by the Applicant to make planned batch releases during meteorological conditions which minimize radiation exposures (in addition to complying with regulations stipulating permissable levels of radiation,

C9-3 (cont.)

radioactivity in effluents, design criteria, and limiting conditions for operation) complies with the requirements of 10CFR§20.1(c).

CQ-4

Interrogatory State the basis for your assertion that there will be "effects of radioactive releases on the general public other than at the exclusion boundary".

Original Answer The BEIR reports, 10CFR Part 20 and 10CFR Part 50 including references cited.

Motion To Compel ... Neither the Staff, the Licensing Board nor the parties should be required to speculate as to the portions of "the BEIR reports, 10 CFR Part 20 and 10 CFR Part 50 including references cited" which contain the basis for CFUR's assertion that there will be "effects of radioactive releases on the general public other than at the exclusion boundary". ...".

Supplementary Answer (5/22/81) There is no reason that CFUR can think of for assuming that there will be effects of radioactive releases on the general public only at the exclusion boundary. CFUR does not know what the Staff is asking.

Telephone Contact 8/18/81 CFUR unable to obtain any further explanation of interrogatory. Staff insists that CFUR should provide page numbers of documents as ordered by Board.

2nd Supplementary Answer 10CFR§20.105 and 20.106 prescribe permissable levels of radiation in unrestricted areas and radioactivity in effluents to unrestricted areas. These requirements apply to all unrestricted areas - not just at the exclusion boudary. With respect to design objectives and limiting conditions for operation, 10CFR§50 Appendix I, Sec. III states that account shall be taken of the cumulative effect of all sources and pathways within the plant contributing to the particular type of effluent considered and that estimation of exposure shall be made with respect to such potential land and water usage and food pathways as could actually exist during the term of plant operation. In addition, 10CFR§50 Appendix I states that the characteristics attributed to a hypothetical receptor for the purpose of estimating internal dose committment shall take into account reasonable deviations of individual habits from the average. These requirements obviously apply to an evaluation ot the effects of radioactive releases on the general public other than at the exclusion boundary (in addition to persons at the exclusion boundary).

Regulatory Guide 1.111 supplies methods for estimating atmospheric transport and dispersion of gaseous effluents in routine releases from light-water-cooled reactors and Regulatory Guide 1.145 supplies atmospheric dispersion models for potential accident consequence assessments at nuclear plants. Neither restrict an evaluation of the effects on the general public to just the exclusion boundary.

Chapter IV of the BEIR I report Section II.B "Atmospheric Dispersion and Removal Processes", Section II.D "Land-based Nuclear Facilities", Section IV.C "Other Sources of Radiocontamination", Section IV.D "Redistribution of Radionuclides", Section IV.E "Radiation Effects In Soil", Section IV.F "Radionuclide Entry into Plants and Radiation Effects", and Section V.C "Animal Products as Sources of Human Exposure" document sufficiently that effects of radioactive releases on the general public occur other than at the the exclusion boundary.

Appendix VI "Calculation of Reactor Accident Consequences" of WASH-1400, Appendix II "Pollutant Pathways" of "Public Health Risks of Thermal Power Plants", UCLA-ENG-7242 by Starr and Greenfield, and "Radioactive Releases From Nuclear Installations", Vol. 2, pp. 17-152, by Clarke and MacDonald all document effects of radioactive releases on the general public other than at the exclusion boundary.

All of these documents (some of which CFUR does not consider conservative) substantiate that there will be "effects of radioactive releases on the general public other than at the exclusion boundary". CFUR is not aware of any proposed basis that there will be effects of radioactive releases on the general public only at the exclusion boundary.

C9-6

Interrogatory Identify the "various transport mechanisms" referred to in the contention.

Original Answer See C9-5. More complete answer provided with direct testimony.

Motion To Compel ...In these interrogatories the Staff is seeking an identification of the "various transport mechanisms" referred to... CFUR's answers are totally unresponsive. ...

Supplementary Answer (5/22/81) Those transport mechanisms detailed in "AIRDOS-EPA: A Computerized Methodology for Estimating Environmental Concentrations and Dose to Man from Airborne Release of Radionuclides", Oak Ridge Nat'l Laboratory, TN, December, 1979.

Telephone Contact 8/18/81 Staff insists supplementary is not specific enough. Requests page numbers so when they obtain the document they will know where to look.

2nd Supplementary Answer Document number is PB80-147838. Modes of exposure include (1)immersion in air containing radionuclides, (2)exposure to ground surfaces contaminated by deposited radionuclides, (3)immersion in contaminated water, (4)inhalation of radionuclides in air, and (5)ingestion of food produced in the area. Atmospheric and terrestial transport models are included on pages 8 thru 30. Methods of calculating radiation doses and intake rates by persons are included on pages 36 thru 54. Terrestial transport input parameters are included on pages 81 thru 100.

The balance of the report contains the introduction, a section on how to use the code and listings of the code and sample runs which may also be helpful.

C4-10

1st Supplementary Answer, CFUR to NRC Staff:

Those of the Lewis Committee. See attached copy of "CFUR's Report on Each Contention" dated 4-10-80, regarding Contention 4.

C4-12

1st Supplementary Answer, CFUR to NRC Staff:
Report of the President's Commission on the Accident at Three Mile Island; those of the German Risk Study Summary, issued August 9, 1979 by the Federal Ministry of Research and Technology in West Germany; NUREG-0642, "A Review of NRC Regulatory Processes and Functions", p.p. 6-2, 8-3, 8-2, and 7-8; letter, Council on Environmental Quality to John Aherne, March 20, 1980. See attached copy of CFUR's Report on Each Contention" dated 4-10-90, regarding Contention 4.

C4-15

1st Supplementary Answer, CFUR to NRC Staff:
Most probably. A hydrogen explosion could conceivably sever both normal and emergency core cooling as well as cause the integrity of the containment structure to be violated if the hydrogen were to explode. This limiting catastrophic event was not adequately addressed in WASH-1400. Yet, it is clear from the TMI accident that the possibility of sufficient accumulation of hydrogen in either the reactor and/or the containment for an explosion to occur is greater than heretofore imagined. As established in the position statement for this contention, TMI's accident occured with less than 500 reactor-years of commercial operation. The TMI-2 accident is a credible accident. It is also a known fact that a hydrogen explosion occured at TMI: "At about 9½ hours into the accident, the hydrogen in the reactor containment building ignited....", IEEE Spectrum, The Technical Blow-by Blow, p. 42.

Depending on the requirements concerning hydrogen venting, two positions are possible: 1.) Hydrogen venting of the primary coolant will be installed. Much larger quantities of hydrogen than that experienced at TMI-2 will occur in the evet of a partial meltdown similar to TMI-2. Enough oxygen is present in the containment building to support combustion. The hydrogen recombiners installed at CPSES would not be able to remove the hydrogen before a spark from either operator action or from automatically controlled equipment ignited the oxygen.

Therefore, this accident sequence should be evaluated. 2.) Hydrogen venting of the primary coolant system will not be installed. In the event of a partial meltdown, such as occured at TMI-2, non-condensible gases in the primary coolant loop present a problem in that restriction or blockage of primary coolant flow may occur. Such a sequence could lead to full meltdown with all the attendant hazards associated with such occurances. Steam and/or hydrogen explosions then present a serious hazard in that overpressurization of the containment may occur.

The amount of hydrogen which escapes to the containment building in the event that restriction or blockage of the primary coolant flow does not take place is directly proportional measure to the size of the break in the primary flow. A break equal to or less than the TMI-2 break is not assured.

The hydrogen explosion that took place at TMI-2 caused a pressure spike of 28 psi. Had more hydrogen excaped due to a larger break, due to more rapid formation or if the hydrogen had been ignited at a later time, it is reasonable to assume that the pressure spike would have been greater. CPSES has hydrogen recombiners installed in the containment building which can be operated remotely from outside the containment. But the design paramenters used for the containment hydrogen monitoring system to be operational do not require this status until 24 hours after the accident. (CPSES/FSAR, p.6.2-83) As noted above, the TMI-2 hydrogen exploded at about 9½ hours into the accident. THis system would have no value in this circumstance. The CPSES recombiners are designed to limit hydrogen concentration to or below four volume percent based on the release model indicated in Regulatory Guide 1.7 dated March 10, 1971. (CPSES/FSAR, p. 6.2-81 and 1A(B)-3). This guide has been revised at least twice (Sept., 1976 and Nov., 1978). Even then, an exception to the guide is taken concerning assumptions for the analysis of hydrogen production and accumulation in the containment based on the "maximum credible accident" (CPSES/FSAR, p. 6.2-104, Figs. 6.2, 5A, 8 and 9). According to the NRC model and assuming total mixture, the volume percent of hydrogen would exceed 4% after 25 days for the release rate of R.G. 1.7. But what would happen in an accident sequence similar to TMI-2, a credible accident? When would the hydrogen recombiners be turned on? By what procedure and according to what indication? At what rate ~~of~~ would hydrogen be formed? At what rate would hydrogen leak into the containment? What

C4-15, cont.

events may take place to ignite localized hydrogen? In the event of a small hydrogen explosion (20 psi) would the recombiners continue to function or would they turn into a source of oxygen to support successive explosions?

CERTIFICATE

I declare (or certify, verify or state) under penalty of perjury that the preceding answers to NRC Staff's interrogatories are true and correct to the best of my knowledge.

Executed on this 3rd day of September, 1981.

Betty Brink

Betty Brink

Respectfully submitted,

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Position: At the time of the TMI-2 accident, such an occurrence was thought to be "incredible." In fact, after the accident, the NRC staff found that:

> "The accident at Three Mile Island Unit 2 involved a sequence of successive failures (i.e. small-break loss-of-coolant accident and failure of emergency core cooling system) more severe than those postulated for the design basis of the plant. Therefore, we conclude that the accident at Three Mile Island was a Class 9 event." Matter of Public Service Electric and Gas Co. (Salem Nuclear Generating Station, Unit 1), Docket 50-272, "NRC staff response to question no. 4 of the Atomic Safety and Licensing Board" at 3 (emphasis added).

CFUR does not agree with this categorization. Instead, CFUR contends that the TMI-2 accident sequence was mis-categorized in the first place - more in line with the finding of the President's Commission on the Accident at Three Mile Island when the found that:

> "...the probability of occurrence of an accident like that at Three Mile Island was high enough, based on WASH 1400, that since there had been more than 400 reactor years of nuclear power plant operation in the United States, such an accident should have been expected during that period." Report of the President's Commission on the Accident at Three Mile Island 32 (1979) (emphasis added).

Clearly, if one accepts this premise, an accident even more serious than Three Mile Island is credible in light of the fact that the TMI-2 accident occurred in less than 500 reactor years of operation. The problem is how to identify those sequences which fall in the credible category. WASH-1400 supplies information upon which to base the sequence. PWR-3 involves a small LOCA with an equivalent diameter of about 1/2 to 2 inches combined with failure of the containment spray injection system followed by containment failure due to overpressure. The median probability assigned this accident sequence by WASH-1400 is 2×10^{-6} per reactor-year. However, the WASH-1400 report was criticized in a reevaluation by H. W. Lewis' Risk Assessment Group initiated by the NRC. The Lewis Group concluded that WASH-1400 failed to emphasize sufficiently the uncertainties involved in the calculation of probability and that the bounds of error on the estimates of accident sequence probabilities were greatly understated. In light of the criticisms of the WASH-1400 study made by the Lewis Committee, the Nuclear Regulatory Commission reexamined its views regarding the WASH-1400 study and made the following statement:

> "The Commission accepts these findings (of the Lewis Committee) and takes the following action: Accident Probabilities: The Commission accepts the Review Group Report's conclusion that absolute values of the risks presented by WASH-1400 should not be used uncritically either in the regulatory process or for public policy purposes and has taken and will continue to take steps to assure that any such use in the past will be corrected as appropriate. In particular, in light of the Review Group conclusions on accident probabilities, the Commission does not regard as reliable the Reactor Safety Study's numerical estimate of the overall risk of reactor accident.

Attachment -I-1

In addition, the ACRS has addressed this issue:

"...the containment design pressure is based on the assumption that core melting will be maintained and that no fuel melting will occur. The containment does not include provisions to cope with a molten core or the heat, hydrogen, and other aspects of an accident in which the whole coremelts."

"The single failure criterion and other failure control design bases should be modified as necessary to encourage consideration of progressive, common cause, and multiple failures arising from a single initiating event."

"Except for a few limited cases considered during the past few years, the staff has been unwilling to investigate potentially signigicant safety matters if they were not identified as part of the 'design basis'. It's consideration of the ramifications of accidents involving degraded safety features performance or other circumstances leading to accident consequences beyond those covered by the 'design basis' was too restrictive, causing both industry and the regulatory staff to be inadequately prepared for anticipated accident circumstances. There has been a salutary change in the NRC Staff views of such matters since the the TMI-2 accident that seems responsive to the need. Future organizational arrangements should assure that this interest will be sustained."

"Accidents beyond the current design bases should be considered in deciding on the future approach to ... design, and to emergency measures."

"...the SER consists primarily of repetitive 'boiler plate' which often tends to obscure and provide little amplification of safety issues. The result is that the SER has become a document of little value to those people responsible for safety reviews of nuclear facilities." NUREG-0642, "A Review of NRC Regulatory Processes and Functions", p. p. 6-2, 8-3, 8-2, and 7-8.

No such salutory change in NRC Staff views is in evidence in this proceeding and 'boiler plate' analysis appears to satisfy the Staff. But the health and safety of the public in the vicinity of CPSES requires something more than this approach.

The Council on Environmental Quality states:

"The past failure to discuss the consequences of the full range of potential accidents and their effects undermines the basic purpose of the National Environmental Policy Act to inform the public and other agencies fully of the potential consequences of Federal proposals and to provide a basis for informed decisions... We do not believe the Commission's prior legal justification for severly limiting the discussion of nuclear accidents and their consequences in EIS's is any longer sustainable, assuming it ever was."
Letter, CEQ to John Ahearne, March 20, 1980.

The following regulation is cited for justification.

"If...the information relevant to adverse impacts is important to the decision and the means to obtain it are not known (e. g. , the means for obtaining it are beyond the state of the art), the agency shall weigh the need for the action against the risk and severity of possible adverse impacts were the action to proceed in the face of the uncertainty. If the agency proceeds, it shall include a worst case analysis and an indication of the probability or improbability of it's occurrence." 40 CFR, Part 1502,22(b) (1979).

The NRC chairman, John Ahearne, in answer to the CEQ, said a staff recommendation to abandon the "old AEC policy" and to discuss serious accidents in environmental impact statements is now before the Commission and would be given prompt consideration. (Inside NRC, March 24, 1980, p. 5). An NRC staff paper (Secy 80-131) advocates, as an interim NRC policy, consideration of core melt events in environmental impact statements and safety reviews. (Inside NRC, April 7, 1980).

In any event, CFUR contends that a PWR-3 accident is a credible accident and the consequences of such an accident should be calculated for CPSES..Even in the event that the probability of the accident cannot be proved, enough uncertainity exists that the accident should be evaluated to meet the "conservative requirements" of 10 CFR , Part 50.

In addition, CFUR contends that an accident sequence based on site-specific initiating events should be analyzed and the consequences determined. The CPSES area is noted for the unusually high frequency and intensity of tornadoes. An accident sequence whereby every designated non-safety function is assumed to be demolished abruptly while both reactors are operating at full load should satisfy this purpose. A combination of tornado-induced missles which initiate additional turbine-generator missles which destroy piping, condensers, and every other so-called nonsafety item could be considered as the initiating event.

With respect to the component parts of the Study, the Commission expects the Staff to make use of them as appropriate, that is, where the data base is adequate and analytical techniques permit. Taking due account of the reservations expressed in the Review Group Report and in its presentation to the Commission, the Commission supports the extended use of probabilistic risk assessment in regulatory decisionmaking." (NRC Statement on Risk Assessment and The Reactor Safety Study Report (WASH-1400) In Light of The Risk Assessment Review Group Report, January 18, 1979.)

Recently, Mr. Lewis noted that WASH-1400 had at least identified the relative importance of various accident types:

"For example, WASH-1400 concluded that transients, small LOCA and human errors are important contributors to overall risk, yet their study is not adequately reflected in the priorites of either the research or regulatory groups. These three items - transients, small loss-of-coolant accidents and human errors - were the central features of the Three Mile Island accident." (H.W. Lewis, "The Safety of Fission Reactors", Scientific American (March 1980), p. 64)

This conlusion is shared in "The German Risk Study Summary" issued August of 79 by the Federal Ministry of Research and Technology in West Germany. The study concludes that 72 percent of all hypothetical core-melt accidents are caused by small reactor pipe breaks. For this kind of accident, about two-thirds of the risk is in human failure and the remainder in equipment failure. One reason why human failures create so much risk is that most postulated accidents come from small reactor leaks such as occurred at TMI, not from large pipe breaks. Large pipe breaks, which empty a lot of reactor water in a hurry, have to be handled promptly and automatically, mostly without operator intervention. This is not so with small pipe breaks.

In light of the above, it is apparent that the probability of a small LOCA is much larger than that used in the WASH-1400 study. In like manner, maintenance error, operator error and /or equipment malfunction could contribute to the probability of failure of the containment spray injection system. Error bounds determined from the Lewis Committee working papers coupled with use of human error rates experienced under stress and the use of a 95% confidence level will establish this probability. Containment failure by overpressure is described as follows:

"According to an NRC source, containments were expected to withstand even core melts until the mid-1960's when the idea became'too expensive' to consider....The limit is now 50 psi but 'with margins' - it can withstand 100 psi. The TMI pressure spike went up to 28 psi. But other kinds of accidents - a steam explosion, for instance - could cause pressure to exceed 100 psi, particularly in a core melt reaction with concrete in which carbon dioxide, steam and hydrogen may be liberated." Inside NRC, Volume 2, No. 7- April 7, 1980, p. 7.

CERTIFICATE OF SERVICE

I certify that a copy of the foregoing document has been forwarded to all parties of record this 3rd day of September, 1981, by deposit in the United States Mail.

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