4/10/80

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

In the Matter of)

TEXAS UTILITIES GENERATING COMPANY, ET AL.) Docket Nos. 50-445
50-446

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

REPORT OF CFUR's POSITION ON EACH CONTENTION

In response to the ASLB "Order Scheduling Prehearing Conference" dated March 19, 1980, a complete report on each of CFUR's Contentions is enclosed (Enclosure 1). A stipulation with an Attachment of "Statement of CFUR Contentions" indicating each parties position with respect to each contention has been forwarded under seperate cover.

One generic issue exists pertaining to CFUR's position regarding contentions 2.B, 3.B and possibly 8. At the July 17, 1979 meeting in Arlington, Texas between the Staff, Applicant, Texas AG and CFUR (the only meeting held in regard to arriving at stipulations), the Staff indicated that it was their opinion that contentions 2.B and 3.B were Three Mile Island related and should be deferred. With no discussion of either wording or admissibility, it was agreed that these contentions would be deferred until some unspecified events had taken place. At that time, CFUR requested of the Staff that it be kept informed of anything which was significant concerning these contentions. CFUR assumes that contention 8 was added to this category when in response to "CFUR's Motion To Add Contention" dated October 31, 1979, the Staff proposed to discuss the additional contention wiht CFUR (at a later date - since no discussion took place in this time frame).

CFUR encountered difficulty in getting the Staff to be specific about what the Staff considered to be significant in regard to these contentions. This difficulty resulted in "CFUR's Response To NRC Status Report And Request For Information" dated February 19, 1980. The "NRC Staff Answer To CFUR And CASE Responses To NRC Staff's Status Report On Proposed Stipulations And Their Requests That The NRC Staff Be Ordered To Provide Documents And Conduct Further Negotiations" dated March 10, 1980 contains the following:

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> "It is the Staff's belief that the particular reports the parties had in mind are the NRC Investigation Into The March 28, 1979 Three Mile Island Accident, by Office of Inspection and Enfcrcement (NUREG-0600, completed in July 1979 and published in August 1979); the Report of the President's Commission on the Accident At Three Mile Island (October 1979), and the report of the Rogovin Special Inquiry Group (Three Mile Island - A Report to the Commissioners and the the Public, issued on January 24, 1980)."

This is the most definitive statement CFUR has obtained as to what the NRC Staff had in mind when they suggested that these contentions be deferred. (Although, as may be noted, this states only what the Staff believes CFUR has in mind.)

On August 25, 1979, CFUR made a request for information to the Staff for In-House reports, the Kemeny report and the Rogovin report(among other items - Enclosure 2). In the Staff reply (Enclosure 3), NUREG-0600 was supplied and it was stated that the Kemeny report is outside NRC but that the Rogovin report would be provided when such reports are available. CFUR obtained a copy of the Kemeny report and Volume I of the Rogovin report was forwarded to CFUR through regular NRC channels. Volume II has not been forwarded to date.

Subsequently, on February 19, 1980 (Enclosure 4), CFUR again requested a copy of Volume II of the Rogovin report. In reply, the Staff stated: "... the report is being printed in final form and a copy will be furnished to you upon issuance."

CFUR finds it difficult to formulate a position in the absence of documents which may be significant to its position. CFUR is not convinced that these documents are the only relevant documents. However, attempts to obtain draft copies of the "Staff Action Plan" are in a similar status. Prehaps CFUR should have pursued the Freedom of Information route in an attempt to obtain access to these materials but such efforts are not timely and are expensive. Prehaps CFUR is presumptuous to expect the Staff to identify significant documents. But if these proceedings are to be held in a timely manner to enable seperately planned fuel loading dates to be met while at the same time ensuring that all parties are on an equal footing insofar as is practicable, CFUR believes that it is correct in bringing these difficulties to the attention of the ASLB.

CFUR has attempted to formulate a position on each of its contentions, including these contentions, as may be noted in Enclosure 1. In the event that the documents requested prove to contain additional significant information pertaining to CFUR's contentions, it is requested that the ASLB allow CFUR to modify its position and possibly its wording.

CFUR prays that the ASLB consider and admit all of CFUR's contentions in this proceeding.

Respectfully submitted,

Richard L. Fouke

Richard L. Fouke
CFUR

Dated at Arlington, Texas
this 10th day of April, 1980

ENCLOSURE 1

REPORT ON EACH OF CFUR's CONTENTIONS

CONTENTION 1

Applicants have not demonstrated technical qualifications to operate CPSES in accordance with 10 CFR part 50.57(a)(4) in that they have relied upon Westinghouse to prepare a portion of the Final Safety Analysis Report (FSAR).

POSITION;

The position of CFUR is adequately stated in "Supplement To Petition For Leave To Intervene By Citizens For Fair Utility Regulation (CFUR)" dated May 7, 1979. The position described under: "I Corrective Actions Necessary Due To Preparation And Defense Of Comanche Peak Final Safety Analysis Report By Westinghouse In Lieu Of Preparation And Defense By Applicant" on pages 1 and 2 of Supplement is adapted as CFUR's position.

CONTENTION 2.A

One or more of the reports used in the construction of computer codes for the CPSES/FSAR have not been suitably verified and formally accepted; thus conclusions based upon these computer codes are invalid.

POSITION:

In order for the Commission to issue an operating license, there must be a finding that:

> "(1) Construction of the facility has been substantially completed, in accordance with...the provisions of the Act, and the rules and regulations of the Commission; and
> (3) There is reasonable assurance (i) that the activities authorized by the operating license can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the regulations in this chapter; and
> (6) The issuance of the license will not be inimical to the common defense and security or to the health and safety of the public." (10 CFR part 50.57(a))

CFUR contends that in the absence of a formal review which specifically determines that all of the applicable reports and/or codes have been suitably verified, the above finding cannot be properly made. The computer codes referred to are the basis for the assertion by the applicant that the facility has been constructed properly and that accident sequences are containable. Actions on the part of the Staff to assure that the material is applicable to the specific plant involved is clearly inadequate for such a finding. The lack of adequate review has caused difficulty in the past: "Vepco is now completing its reanalysis of the seismic restraints for the Surry - 2 unit's piping using a computer code without outmoded algebraic summation. A year ago, NRC called for a shutdown of five units that had used outmoded algebraic summation in stress calculations for pipe supports."[1]

It was clearly improper to have concluded that these plants met the criteria of 10 CFR Part 50.57(a).

Section 1.5, 1.6 and 3.7B(A) of the CPSES/FSAR documents the status of Requirements For Further Technical Information, Material Incorporated By Reference and Computer Programs Used in Dynamic and Static Analyses. In Section 1.6 of the CPSES/FSAR, the status of the material is listed. A large number of references are listed as being "Actively Under Formal NRC Review." Among those so listed is THINC-IV, WCAP-7956. The Staff supplied CFUR with a copy of an NRC letter dated April 18, 1978 which states:

> "As a result of our review of WCAP-7956 and WCAP-8056, we have concluded that the THINC-IV computer code is acceptable for performing steady-state hydraulic calculations in reactor cores provided suitably conservative assumptions are used with respect to plant operating conditions, fuel fabrication tolerances, and power peaking uncertainties. Fluid conditions are limited to the single phase or the homogeneous two phase flow regime. Limitations on the use of the THINC-IV are provided and fully discussed in our evalution of these topical reports (Enclosure 1)."

Enclosure 1 was not supplied to CFUR. The means of verification of the code was likewise not supplied. Apparently, the Staff is of the opinion that this code has been formally accepted by the NRC, but the applicant does not (through amendment 7 of the CPSES/FSAR). Due to the limited information available to CFUR, a firm decision as to the status of this particular code is not possible.

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[1]Inside NRC, Volume 2, No. 7, April 7, 1980, p.13

CFUR's contention is that one or more of the reports used in the construction of computer codes for the CPSES/FSAR have not been suitably verified and formally accepted. The THINC-IV code is only one of many in this status (according to the applicant). In addition to the many Westinghouse reports which have not completed formal review, there are some Gibbs and Hill codes with questionable status. Among the latter are the SCONV and SPECTRA codes. No reference documentation of the SCONV is supplied in the CPSES/FSAR. No reference documentation nor indication as to how the A/E allegedly verified the SPECTRA code is supplied in the CPSES/FSAR. The MRI/Stardyne code is listed as being available in thepublic domain in the table accompanying Section 3.7B(A) but the reference indicates that it was developed and proprietary to Mechanics Research, Inc. No indication as to the status of NRC review and verification was supplied.

A more complete listing of questionable reports and codes is included in Tables 1 and 2 .

It may be assumed that the Staff will independently review all of material in question and in each case prove that suitable verification has been attained. But there is no basis for this assumption. Much of the Westinghouse material is dated material. It is difficult to believe that at least some of the material still under formal NRC review has not been referenced in past operating license proceedings. If so, there certainly is no justification for assuming that anything will be different this time around.

This is the first nuclear plant that Gibbs and Hill has been involved with as an A/E. The need for stringent formal review and verification is apparent, especially in viewof the Staff's previously mentioned experience with another A/E.

This contention is obviously a proper issue for this proceeding.

TABLE 1.

Partial List of Material Used Which Has Not Been Approved by the NRC

1. "An Evaluation of Solid State Logic Reactor Protection in Anticipated Transients," WCAP-7706-L (Proprietary) and WCAP-7706 (Non-Proprietary), February 1973

2. "Overpressure Protection for Westinghouse Pressurized Water Reactors," WCAP-7769, October 1971

3. "Overpressure Protection for Westinghouse Pressurized Water Reactors," WCAP-7769, Revision 1, June 1972

4. "Evaluation of Steam Generator Tube, Tubesheet and Divider Plate Under Combined LOCA Plus SSE Conditions," WCAP-7832, December 1973

5. "LOFTRAN Code Description," WCAP-7907, June 1972

6. "FACTRAN - A FORTRAN-IV Code for Thermal Transients in a UO_2 Fuel Rod," WCAP-7908, June 1972

7. "MARVEL, A Digital Computer Code for Transient Analysis of a Multiloop PWR System," WCAP-7909, June 1972

8. "Fuel Assembly Safety Analysis for Combined Seismic and Loss of Coolant Accident," WCAP-7950, July 1972

9. "Reactor Coolant Pump Integrity in LOCA," WCAP-8163, September 1973

10. "Safety Analysis of the 17x17 Fuel Assembly for Combined Seismic and Loss of Coolant Accident," WCAP-8236 (Proprietary), December 1973 and WCAP-8288 (Non-Proprietary), January 1974

11. "Documentation of Selected Westinghouse Structural Anlaysis Computer Codes," WCAP-8252, Revision 1, July 1977

12. "Hydraulic Flow Test of the 17x17 Fuel Assembly," WCAP-8278 (Proprietary) and WCAP-8279 (Non-Proprietary), February 1974

13. "Westinghouse Anticipated Transients Without Trip Analysis," WCAP-8330, August 1974

14. "An Evaluation of Loss of Flow Accidents Caused by Power System Frequency Transients in Westinghouse PWRs," WCAP-8424, Revision 1, June 1975

15. "Fuel Rod Bowing," WCAP-8691 (Proprietary) and WCAP-8692 (Non-Proprietary), December 1975

16. "Improved Analytical Models Used in Westinghouse Fuel Rod Design Computations," WCAP-8720 (Proprietary) and WCAP-8785 (Non-Proprietary), October 1976

17. "Westinghouse Emergency Core Cooling System Evaluation Model - Modified October 1975 Version," WCAP-9168 (Proprietary) and WCAP-9169 (Non-Proprietary), September 1977

18. "Properties of Fuel and Core Component Materials," WCAP-9179 (Proprietary). September 1977

19. "MRI/STARDYNE, Static and Dynamic Structural Analysis System, User's Information Manual," developed by and proprietary to Mechanics Research, Inc.

20. "SCONV", ref. cited in CPSES/FSAR 3.7B(A).9, p. 3.7B(A)-12

21. "SPECTRA", ref. cited in CPSES/FSAR 3.7B(A).10, p. 3.7B(A)-15

22. "Safety-Related Research and Development for Westinghouse Pressurized Water Reactors, Program Summaries, " WCAP-8768, latest revision

TABLE 2.

Cross-Listing of Materials by Usage and Area Affected

	Justification	Support	Model
1.5 Requirements for Further Technical Information			22
3.6N Protection Against Dynamic Effects Associated with the Postulated Rupture of Piping			11
3.7B(A) Computer Programs Used in Dynamic and Static Analyses			19,20,21
3.7N Seismic Design	8	10	
4.2 Fuel Rod Performance	18	10	15,16
4.3 Nuclear Design		13	
4.4 Thermal and Hydraulic Design		12	
4.6 Functional Design of Reactivity Control Systems		1	
5.2 Integrity of Reactor Coolant Pressure Boundary	3, 5	3	
5.4 Component and System Design		4, 9	
6.3 Emergency Core Cooling System			7

	Justification	Support	Model
15.1 Increase in Heat Removal By Secondary System			5, 13
15.2 Decrease in Heat Removal by Secondary System			2, 5, 6, 13
15.3 Decrease in Reactor Coolant System Flow Rate			5, 6, 14
15.5 Increase in Reactor Coolant Inventory			5
15.6 Decrease in Reactor Coolant Inventory			5, 17

Contention 2.B

The computer codes used in CPSES/FSAR must be tested and, if necessary, modified to accept the parameters reflecting the sequence of events at Three Mile Island and then used to realistically predict the behavior observed at Three Mile Island in consideration of these parameters.

Position: The Three Mile Island accident occureed with less than 500 reactor year experience. The probability of occurrence for this type of accident is not so low as to be considered incredible. From 10 CFR Part 50, "Protection Against Accidents In Nuclear Power Reactors":

> "In the approach to safety reflected in the Commission's regulations, postulated accidents, for purposes of analysis, are divided into two categories - 'credible' and 'incredible'. The former ('credible') are considered to be within the category of design basis accidents. Protective measures are required and provided for all those postulated accidents falling within that category,...Those accidents falling within the 'incredible' category are considered to be so improbable that no such protective measures are required.
> The application of Parts 50 and 100 helps assure public safety by basing protection to the public, both in design and siting, on a very conservative standard for determining and calculating the consequences of postulated potentially severe 'credible' accidents"

The actions of the NRC since the occurrence of TMI-2 also leads one to conclude that TMI-2 was a credible accident. Many protective measures have been required of nuclear power plant operators as a direct result of TMI-2 (TMI-2 Lessons Learned Task Force and I and E bulletins on small-break LOCAs and loss of feedwater accidents) and many more are contemplated (Staff Action Plant). As may be noted above from 10 CFR 50, credible accident protective measures are required and provided for while incredible accidents are so improbable that no such protective measures are required. One is left with no other choice than to conclude that TMI-2 was a credible accident. It follows that, in order to comply with 10 CFR 50, protection of the public must be based on a very conservative standard for determining and calculating the consequences of postulated potentially severe TMI-2 type accidents.

TMI-2 Accident Description

The accident involved three different elements:

1. Maintenance Error

A number of maintenance activities were intertwined with the TMI-2 accident. The actions of the shift foreman and two auxiliary operators in an attempt to unclog a resin plug in a pipe leading from a condensate polisher initiated the sequence of events. Prior maintenance activities also were contributing factors in the accident. The feedwater valves which were improperly left closed are obvious contributors. But in addition to this action, sluggish response on the part of maintenance to cure problems with instrumentation in the control room was also a contributor, especially when it is recognized that maintenance tags obscured the view of some of the instruments which were operating normally.

2. Equipment Failure

The Pilot- Operated Relief Valve performed normally when it opened, which is its primary function. However, the valve failed to close properly, which will be referred to as its secondary function.

3. Operator Error

A number of significant operator errors escalated conditions at TMI-2. Besides not being trained for this accident sequence the material available to them had apparently convinced them that the sequence of events which occurred at TMI-2 were impossible. The number of errors are legion, complex, and would take too long to enumerate. A certain contributor to these errors was the fact that the FSAR was deficient but never challenged in the regulatory process:

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

<u>Inadequacy of the Computer Code Accident Sequences</u>

An examination of faults, failure to assume maintencance error, equipment failures of a secondary nature, and operator errors during the course of a fault renders the conclusions reached invalid.

Maintenance error could disable systems counted on to avoid or mitigate serious conditions. Such errors are a relatively common occurrence, particularly the closing of feedwater valves. Such errors could lead to serious financial and ecological consequences. Failure of a component performing a secondary function could lead to a complex fault requiring precise operator action to avoid serious ramifications. The failure of the PORV at TMI-2 is a common enough occurence to warrant comment in NUREG-0460, ATWS, Vol. 4:

> "Experience with operating plants has shown that PORV"s may be isolated (sometimes for long periods of operation) due to problems of leakage through valves." (U.S.N.R.C., NUREG-0460, Anticipated Transients Without Scram for Light Water Reactors, March, 1980, Appendix D, P.2)

In addition, the PORV valve in a Westinghouse reactor caused events similar to those experienced at TMI-2 in that a solid pressurizer indicating that the RCS was full of water instead of the situation:

> "Virtually identical transients, as they are called in industry, occurred in 1974 at a Westinghouse reactor in Beznau, Switzerland and in 1977 at Toledo Edison's Davis-Besse plant in Ohio,...Both involved the same failed-open pressurizer relief valve(PORV), and the same misleading indications to operators that the reactor coolant system was full of water....The NRC never learned about the incident at the Beznau reactor until after the TMI-2 accident (March 28, 1979)...but neither the Beznau incident nor the earlier study (of this kind of incident) had prompted Westinghouse to notify its customers or the NRC that operators might well be misled by their instruments if a valve stuck open." (USNRC, "Special Inquiry Group, Three Mile Island: A Report to the Commisioners and To The Public", Vol.1, Jan, 1980, p. 94)

Coupled with such maintenance errors as inadvertent closing of feedwater valves, equipment failures of a secondary nauture tend to eliminate options for the reactor operator.

Operator actions,other than those that initiate the fault conditions, are assumed to be correct through analyses of fault sequences. However, operators incur high stress conditions during faults that lead to operator errors. Proper actions are not always completed and/or improper actions are taken, due to the failure to discern or discriminate among multiple inputs. It is difficult to establish priority actions when stress blocks informational flows and inhibits organization of thought and action.

Maintenance error, equipment failure and operator error occurred at TMI-2 despite the accumulation of reactor experience and programs designed to minimize such mistakes. Without incorporation of maintenance, equipment failure and operator action in computer codes designed to model fault behavior, the conclusions reached cannot be construed as wholly adequate. Without analysis of interactive effects on multiple or common mode failures, procedures stemming from computer codes are not indicative of necessary mitigative sequences or activities.

The basic assumption of the Staff argument against this contention is that the variation in vendor systems between the Babcock and Wilcox - TMI system and the Westinghouse - CPSES system makes comparison and contrast of these systems a pointless exercise. The assurance that a TMI-style transient or accident could happen at CPSES merely on the basis of vendor system variations is to ignore the fundamental impacts of relevant factors.

It has not been documented to this intervenor's knowledge or satisfaction, that the configuration and components used at Comanche Peak result in a sufficiently more reliable situation as compared to TMI-2 to warrant ignoring equipment failures in secondary modes when analyzing the design basis accidents. Maintenance error and operator error compound the consequences of postulated accident sequences.

When a particular maintenance and/or operator error has been deemed to be of primary importance, such as those which occurred at TMI-2, it is feasible that training programs and alertness may significantly alter the probability of reoccurrence in a short time frame. But, over the long term (40 years according to operating license request - or to year 2023), and for errors other than those deemed important but leading to serious consequences if ignored, the probabilities remained undefined and and unaccounted for.

In light of the impact of TMI with regard to maintenance and operator error, and the failure of equipment in their secondary function, CFUR contends that the computer codes used in the CPSES/FSAR should be tested and modified to incorporate these fault factors. The behavior of nuclear systems must be modeled to reflect TMI-style transients, after proper deference to vendor variation. All of the accident sequences considered, in addition to the TMI sequence, should take into consideration these factors.

Contention 3.A

Some accident sequences heretofore considered to have probabilities so low as to be considered incredible based, in part, upon the findings of WASH-1400, are in fact more probable in light of additional findings, such as those of the Lewis Committee, and should be evaluated as credible accidents for CPSES. In order to insure conservatism, the probabilities associated with such accident sequences should be the highest probabilities within the specified confidence band.

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 Lesis Committee) and takes the following action: Accedent 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.
> 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 becam '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.

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 cor melts."

> "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 to 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 approaches 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 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 on 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.

Procedure

After the July 17, 1979 meeting this contention was written up by the Staff from notes taken from a spur-of-the-moment description of CFUR's contention which seemed to be a better description of CFUR's concerns. When forwarded, it was obvious to CFUR that some qualifying phrases had beeen left out. It never has been the intent of CFUR to limit the discussion of this contention to WASH-1400 and/or the Lewis Committee report. The Staff was notified of this at the first opportunity and the Staff agreed to the changes as relected in Alternative 2. The Staff also indicated that the Applicant had agree to the changes. In any event, the proper wording of this contention is reflected in Alternative 2.

CONTENTION 3B

A hydrogen explosion accident sequence needs to be added to the list of possible accident sequences for which consequences will be determined for CPSES.

POSITION:

Lack of information makes this contention difficult to discuss. At the time this contention was formulated (May 7, 1979), it was assumed that measures would be taken to vent hydrogen from susceptible locations to avoid the "bubble" effect, thereby restricting the flow of primary coolant water. It is CFUR's understanding that the Applicant has no plans in place to accomplish this, but that the NRC is still contemplating this requirement. As established in the position statement for contention 2.B., Three Mile Island happened with less than 500 reactor-years of commercial operation. The actions of the NRC are consistent with actions expected to be taken for a credible accident. The actions taken are not consistent for an incredible accident. The TMI-2 accident is a credible accident.

It is also a known fact that a hydrogen explosion occurred 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.

This fact was not recognized at the time of occurrence, in part, due to the negligence of the NRC regulatory function in allowing FSAR's to be deficient:

"It seemed to everybody in the control room, whether they were NRC or GPU or B&W, they all came to the same conclusion: that there wasn't anything particularly significant about that spike.

Haynes: That it was not due to a pressure spike? It was due to an electronic signal or ...electrical transient? That was your evaluation?

F: Sure...What type of transient can cause a two million cubic foot building to pressurize and depressurize that quickly?

H: I thought we were talking about the instruments.

F: That's why none of us considered it plausible. It's impossible to do that.

H: I wouldn't say it was impossible. I thought it actually occurred.

F: Based on our training, it was impossible...If you look back through everybody's training and the FSAR and safety analysis and the building construction, you will not see a paragraph that pr ojects that type of transient. (It) is so particularly foreign and unbelieveable that it has absolutely no significance. That's why nobody did anything about it for two days."

(Rogovin report, p. 43)

Depending on the requirements concerning hydrogen venting, two positions are possible:

Postion 1: It is assumed that hydrogen venting of the primary coolant system will be installed. Much larger quantities of hydrogen than that experienced at TMI-2 will occur in the event 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 hydrogen. Therefore, this accident sequence should be evaluated.

Position 2: It is assumed that hydrogen venting of the primary coolant system will not be installed. In the event of a partial meltdown, such as occurred 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 occurrences. 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 doe s not take place in directly proportional measure to the size of the break in primary flow. A break equal to or less than the TMI-2 break is not assurred.

The hydrogen explosion that took place at TMI-2 caused a pressure spike of 28 psi. Had more hydrogen escaped 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 and can be operated remotely from outside the containment. But the design parameters 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 mixing, 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 would hydrogen be formed? At what rate would hydrogen leak into the containment? What events may take place to ignite localized hydrogen? In the event of a small hydrogen explosion (28 psi) would the recombiners continue to function or would they turn into a source of oxygen to support successive explosions? This intervenor contends that this is certainly a proper subject for this proceeding.

CONTENTION 4. A

The Applicants have failed to establish and execute a quality assurance/quality control program which adheres to the criteria in 10 CFR, Part 50, Appendix B.

POSITION:

This contention accurately reflects some of CFUR's concerns. The position of CFUR is adequately stated in "Supplement To Petition For Leave To Intervene By Citizens' For Fair Utility Regulation (CFUR)" dated May 7, 1979. The position described under :"IV. A Lack of Organization, Quality Assurance, etc, IV. B Welding, IV. C Steel, IV. E Concrete, IV. F Expansion Joint and IV. G Fracture Toughness Testing" on pages 4 thru 17 is adapted as CFUR's position.

CONTENTION 4. B

Applicants have failed to demonstrate sufficient managerial and administrative controls to assure safe operation as required in 10 CFR, Part 50, Appendix B. Therefore, special operating conditions should be required.

POSITION:

CFUR contends that there is a need for unusual conditions to be adopted as a prerequisite for an operating license in view of past lax managerial and administrative controls. The main thrust of CFUR's position is contained in "Supplement to Petition for Leave to Intervene by Citizens' for Fair Utility Regulation (CFUR)" dated May 7, 1979. This portion of CFUR"s position is described under: "IV. H Need for Unusual Conditions as a Prerequisite for an Operating License" on pages 17 and 18.

In addition, considerable uncertainity has evolved concerning parameters commonly used in environmental radiological assessment models. Some attempts to evaluate the uncertainity have been made, but there is an inadequate data base for generic studies and almost none for the CPSES site area. It is generally recognized that a minimum of 30 data points is needed before a probability density function for a given parameter can be represented. But few of the parameters analyzed in a report using as much data as available (Hoffman, et al., "An Evaluation of Uncertainities in Radioecological Models", Oak Ridge National Laboratory) had a data base which met this minimum requirement.

CFUR's contention that the outer perimeter of the CPSES site be dedicated to supplying a portion of the Applicants management food chain would also be useful in accumulating site-specific data for these data. This practice would also serve as a health precaution - at least for those aspects which are currently measurable. Such an addition to the objective would introduce financial burdens which, if passed on to the customer, may be considered to be unfair. This addition, however, is actually an R&D addition and federal tax monies should be available for this type of practice.

CFUR contends that this is a proper subject for this proceeding.

CONTENTION 5

There is no assurance that the Spent Fuel Pool area can withstand the effects of tornadoes, as required by 10 CFR Part 50, Appendix A, Criterion 2 because:

a. The analyses upon which the Design Basis Tornado (DBT) is based are perfunctory, outdated and unreliable;

b. The loading analyses based on the Design Basis Tornado (DBT) are inappropriate because they fail to consider the potential loading combination of the DBT and a tornado-generated missile;

c. The assignment of a loading factor of 1.0 for load combination equations incorporating tornado loadings in combination with "normal and accident conditions" is unacceptable;

d. The DBT parameters used in FSAR Section 3.3.2.1 are less conservative than the parameters found in NRC Regulatory Guid 1.76 c.2.

Position:

The position of CFUR os adequately stated in "Supplement, etc." dated May 7, 1979. The position described under: "V Question of Capability of Spent Fuel Stotage Area to Withstand Tornadoes" on page 19 thru 22 is adapted as CFUR's position.

CFUR requested that it be allowed substitute "Category I structures" instead of "Spent Fuel Storage Area" in the wording of this contention. The Applicant objected and CFUR has not done so. But if the basis for this contention is read there is certainly no justification for limiting discussion to the Spent Fuel Staorage area. In the event that this contention is admitted, it would add more to the record to include all Category I structures and it is suggested that the wording be changed to allow this.

CONTENTION 6

Applicants have failed to adequately evaluate whether the roak "overbreak" and subsequent fissure repair using concrete grout have impaired the ability of Category I structures to withstand seismic disturbances.

Position:

The position of CFUR is adequately stated in "Supplement, etc." dated May 7, 1979. The position described under: "VI Overexcavation" on page 22 is adapted as CFUR's position..

CONTENTION 7

Applicants have failed to adequately evaluate the impacts of the drawdown of the groundwater under CPSES during and as a result of plant operation.

Position:

The position of CFUR is adequately stated in "Supplement, etc." dated May 7, 1979. The position described under: "IV. D Groundwater Withdrawal" on pages 12 and 13 is adapted as CFUR's position.

CONTENTION 8

Applicants have failed to make any effort to determine the effect of radioactive releases on the general public other than at the exclusion boundary. Various transport mechanisms may cause, in certain cases, the bulk of the health effects to occur some distance from the exclusion boundary.

Position:

As explained, the wording of this contention has not really been discussed. CFUR has noted that the Staff seemed to be surprised when the object of the contention was explained to them in verbal form. For this reason, CFUR has requested that this contention be considered deferred, since a change in wording may be necessary in order for the contention to reflect CFUR's concerns. Two actions would take place in the event that this contention is adopted. One would be that batch releases of radioactive gases from CPSES would be timed according to meteorlogical criteria (insofar as is possible) such that the cumulative man-rem of exposure of the public would be minimized in accordance with the objective of the ALARA criteria of 10 CFR Part 50. In order to accomplish this the population density of quadrants around CPSES along with current meterlogical conditions would have to be taken into consideration. The action action is that emergency plans for populations in excess of 50 miles distance from CPSES would have to be in place. These plans may include a means of notifying the public to stay indoors and, possibly, to take thyroid blocking agents. The need for these plans to be put into effect would be determined by an evaluation of the possible cumulative dose committment in the event that met logical conditions at the time of an unplanned release (e. g., wind blowing toward Dallas, Houston or Austin) dictated such action.

The draft report to the Council on Environmental Quality, authored by Jan Beyea of Princeton University, titled "Some Long Term Consequences of Hypothetical Major Releases of Radioactivity to the Atmosphere from Three Mile Island" adequately documents that the man-rem doses of releases from nuclear power plants occurs in areas of high density population some distance from the plant in the event that the wind is blowing in that direction. This would still be the situation independent of the magnitude of release. Therefore, it is necessary to take into account this factor when making batch releases.

In addition, meteorlogical data presented in the FSAR does not properly consider the vertical dispersion and transport of radionuclides via storm cloud formations with ultimate localized depositons via rain. Local DFW weather reports have incorporated a time sequence display of weather radar images whenever rain is expected in the Dallas-Fort Worth area. It is apparent that the majority of these movements occur from the southwest to the northwest, which means that the transport mechanisms would cause the radionuclides to travel from CPSES to the DFW area in a majority of cases when releases are make under this condition. It is apparent that more sophisticated weather data is requried to minimize the effects of gaseous batch releases from CPSES in conformance with the ALARA requirements of 10 CFR Part 50.

CFUR contends that this contention addresses subjects pertinent to this proceeding.

CONTENTION 9

The Applicants should be bound to any hardware modifications required to mitigate the consequences of Anticipated Transients Without Scram concerning Westinghouse reactors of the CPSES category even if the Commission grants an exemption to Applicants based upon some specific time frame.

Position:

At first glance, this contention may appear to be challenging the Commission's authority. However, that is not CFUR's intent. Rather, it is intended to have enough action taken so that the dilatory practices of this particular Applicant would be brought to the attention of the Commission prior to any Commission decision concerning ATWS for CPSES. It is suggested that this requirement be made a condition for licensing and then the Commission could change that condition if it so chooses.

The ATWS safety problem is clearly identified at the construction license phase. In the NRC safety evaluation of Commanche Peak, the NRC staff assumed the following position:

> "In it's letter dated September 23, 1974, the Applicant has documented the information required by WASH-1270 by referencing Westinghouse topical reports WCAP-8330, 'Westinghouse ATWT Analysis', and WCAP-7706, 'An Evaluation of Solid State Logic Reactor Protection in Anticipated Transients'. The applicant states that no hardware modifications are required to mitigate the consequences of an ATWS.
> We are reviewing this Westinghouse information on an generic basis, and when the results of our generic review are available, we will evaluate their applicability to the Comanche Peak facility." 1

1 U. S. A. E. C., "Supplement to Safety Evaluation of the Comanche Peak Steam Electric Station, Units 1 and 2", Docket Nos. 50-445-446, November 15, 1974, p.-9.

The ACRS also referred to the ATWS matter in it's report dated October 18, 1974, which referrenced it's report dated February 13, 1974. The ACRS commented: "These problems should be dealt with appropriately by the Regulatory Staff and the Applicant." (2) The ACRS has defined the responsibility of the Applicant in the following manner:

> "The Applicant is the designated license holder under NRC rules. He has to show both financial and technical competence...The Plant owner..is responsible to the NRC for...providing appropriate engineered safety features for the system...and for providing a competent organization to design, construct and operate the plant. Normally, an owner can satisfy only a portion of this capability with his own organization. The remainder is provided through contract agreements with other organizations. Nevertheless, the owner is ultimately held responsible by the NRC for the safety of his plant." (3)

The Applicant has, nevertheless, refused to admit that ATWS is a safety matter, as indicated in Reference 1. Furthermore, TUGCO has been totally negligent in regard to resolution of ATWS safety issues, citing the following concerning ATWS in the FSAR:

> "A discussion of ATWS is presented in Reference (1). The information provided in Reference (1) is applicable to the Comanche Peak Steam Electric Station."

The only reference cited was WCAP-8330.

A significant amount of correspondence has been forwarded by Westinghouse conderning ATWS since the date (1974) of the only reference supplied by TUGCO in the FSAR. These include:

1. Letter from C. Eicheldinger, Westinghouse, to J. F. Stolz, NRC, enclosing NS-CE-1583, attachment 2, October, 1977.

2. Letter from T. M Anderson, Westinghouse, to R. J. Mattson, NRC, with enclosure dated June 8, 1979.

3. Letter from T. M. Anderson, Westinghouse, to S. H. Hanauer, NRC, with enclosure dated December 30, 1979.

4. Westinghouse Electric Corporation, "TRANFLO Steam Generator Code Description", Westinghouse Report WCAP-8821, Sept., 1976.

5. Westinghouse Electric Corporation, "LOFTRAN Code Description", Westinghouse Report WCAP-7878, Revision 7, January, 1977.

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2. Ibid, p. B-3.
3. USNRC, ACRS, "A Review of NRC Regulatory Processes and Functions", NUREG-0642, January, 1980, p. 5-1.
4. TUGCO, CPSES-FSAR, thru Amendment 7, p. 15.8-1.

According to 10 CFR Part 50.34(b) the Final Safety Analysis Report shall include the following:

> "For nuclear reactors, such items as the...instrumentation and control systems...other engineered safety features, auxiliary and emergency systems...shall be discussed insofar as they are pertinent."
> "(4) A final analysis and evaluation of the design and performance of structures, systems and components with the objective stated in paragraph (a)(4) of this setion and taking into account any pertinent information developed since the submittal of the PSAR."

Had TUGCO considered ATWS as a safety matter, it could have been expected to participate in the resolution of this problem and could have undertaken to change the course or nature of such resolution, if so motivated. TUGCO has not done so and has waited about 6 years to reiterate that, in its judgement, "No hardware modifications are required to mitigate the consequences of ATWS", without any revision of the basis for such a judgement.

When one examines the status of ATWS in 1980, it becomes abundantly clear that substantial hardware modifications and analyses are required to insure the health and safety of the public, including those in proximity to CPSES. This is indicated by the following quotes from NUREG-0460, Volume 4, published in March, 1980:

> "The staff has not completed the review of the TRANFLO code... verification of the code (LOFTRAN) is not complete...the system transient code, LOFTRAN, is an inappropiate code to predict system behavior after the formation of voids in the primary loop... no analysis was provided assuring both PORV's isolated. This is a credible situation which actually exists is some operating plants.. Westinghouse did not consider the potential effect of the voids generated in the primary system during the stuck-open pressurizer safety/relief valve event or as a consequence of another ATWS event, These voids may prevent natural circulation of the primary coolant... Westinghouse did not consider the effect of different HPSI design on ATWS. This could have an important impact on the cause of some transients such as a stuck-open PORV..."

The staff concluded the following: (summarized by CFUR)

(1) The stuck-open PORV was not correctly analyzed.
(2) The long-term shutdown was not adequately addressed, given the impact of voids in the primary system, timing of the reactor coolant pump trip and considering plants with a low-high pressure Safety Injection head.
(3) The ability of instrumentation and equipment needed for safe shutdown to withstand the pressure peak was not addressed at all by Westinghouse.

(4) The impact of isolated PORV's was not adequately addressed.
(5) No stress evaluation is provided for the BOP components.
(6) Questions remain on radiological evaluations with regard to containment isolation.
(7) Design information on mitigating systems was inadequately addressed.
(8) Operability and integrity of the HPSI is questionable on early actuation while primary system pressure is above HPSI design pressure.
(9) The integrity and performance of safety and relief valves has not been assured and the TMI-2 industry studies will not address the problem. (5)

In addition, the NRC staff found in NUREG-0460, Vol. 2, that "Westinghouse's analysis assumes that there is automatic initiation of the turbine trip, automatic initiation of the auxiliary feedwater system, and automatic initiation of containment isolation. However, the staff notes that for Westinghouse plants the systems which initiate turbine trip, auxiliary feedwater, and containment isolation are not diverse from the reactor protection system. Consequently, a common mode failure of the reactor protection system could lead to loss of these three functional systems. Without this diversity and automatic turbine trip, the primary system peak pressures will exceed the design limits, late initiation of auxiliary feedwater may lead to core melt..." (6)

In view, of the NRCs staff current position, as of March, 1980, it is reasonable to conclude that ATWS is a safety matter. Ignorance and/or denial of this issue by the Applicant constitutes gross negligence by said Applicant. The FSAR is obviously incomplete and requries immediate revision in order to reflect current status.

The ATWS problem has been identified by the NRC as safety problem some eleven years ago (7). It was identified at the construction license phase as a safety problem. Yet, in the review of the FSAR by the NRR staff, not not single question regarding ATWS has been asked of the Applicant. Yet the Applicant is ultimately responsible for safety as the entity charged

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5. USNRC, "ATWS, Vol. 4", NUREG--460, March, 1980, P. P. 7-9.
6. USNRC, "ATWS, Vol. 2", NUREG-0460, April, 1978, App. XVII, P. P. 57-59.
7. USNRC, "Inside NRC", March 24, 1980, p. 14.

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with making modifications, training personnel and insuring safe operation in order to mitigate ATWS impacts. The NRR staff has failed to establish technical competence with regard to the Applicant. Moreover, they have failed to adequately design the scope and schedule of analytical efforts as to permit the accomplishment of stated ATWS objectives on a timely basis for CPSES.

Upon review of the Applicant's position, it becomes obvious that the NRC and the Applicant are at odds concerning ATWS. It is the purpose of regulatory proceedings to resolve divergent viewpoints. To ignore them in these proceedings would be negligent itself.

The Division of Systems Safety has devised a series of alternatives for proper corrective actions. The action applicable to CPSES is called Alternative 3A, requiring the incorporation of ATWS mitigating system actuation circuitry (satisfying criteria in Appendix C, Vol. 3, NUREG-0460), a Mss that is diverse and independent from the RPS, while meeting IEEE-279 standards and backing up portions of current trip systems, provisions to close Containment Isolation valves if fuel failure should occur and provision for instruments nec essary for shutdown that withstand peak pressures. (8)

Failure to incorporate ATWS requirements above would lead to diminished mitigation capability and possible core melt, according to the DSS in NUREG-0460, Vol. 4. A core melt is unquestionably hazardous to the health and safety of the public, especially in those areas close in proxmity to CPSES. Since DSS proposes to implement ATWS requirements on a case-by-case basis, CFUR contends that the Applicant should be bound to any hardware modifications required to mitigate ATWS consequences concerning the Westinghouse reactors used at CPSES. This should apply even if the Commission grants an exemption to Applicants based upon some specific time frame. To do otherwise is to ignore the requirements for acting in an "timely basis" in fulfillment of 10 CFR Part 50.40(a).

Lastly, CFUR calls attention to the ASLB the following exercpt from the Kemeny Commission report:

> "NRC's primary focus is on licensing and insufficient attention has been paid to the ongoing process of assuring nuclear safety. An important example of this is the case of 'generic problems', that is, problems which apply to a number of different nuclear power plants. Once an issue is labelled 'generic', the individual plant being licensed is not responsible for resolving the issue prior to licensing. That, in inself, would be acceptable, if there were a strict procedure within NRC to assure the timely resolution of generic problems, either by it's own research staff or by the utility and it's suppliers. However, the evidence indicates that labeling a problem as 'generic' may provide an convenient way of postponing decision on a difficult question."

To ignore this question at the operating license stage is to put the health and safety of the public contingent to CPSES in extreme and unwarranted jeopardy as a result of the temerity of the NRC, the stubbornness of the Applicant, and the neglect of the ASLB.

*8. NUREG-0460, Vol. 4, March, 1980, DSS.

9. John G. Kemeny, Chairman, "Report of the President's Commission on the Accident at Three Mile Island", October, 1979, p. 20.