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NUCLEAR REGULATORY COMMISSION

COMMISSION MEETING

In the Matter of: PRESENTATION BY GE, AIF, AND G. LELLOUCHE (EPRI) ON ANTICIPATED TRANSIENTS WITHOUT SCRAM (ATWS)

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UNITED STATES OF AMERICA

NUCLEAR REGULATORY COMMISSION

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Presentation by GE, AIF, and G. Lellouche (EPRI) on Anticipated Transients without Scram (ATWS)

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Nuclear Regulatory Commission
Room 1130
1717 H Street, N.W.
Washington, D.C.

Tuesday, October 28, 1980

The Commission met at 2:05 p.m., pursuant to notice.

PRESENT:

John Ahearne, Chairman
Joseph Hendrie, Commissioner
Victor Gilinsky, Commissioner
Peter Bradford, Commissioner

PRESENT FOR THE NRC STAFF:

Samuel Chilk

PRESENT FOR THE OFFICE OF GENERAL COUNSEL:

Leonard Bickwit
Marty Malsch

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P R O C E E D I N G S

CHAIRMAN AHEARNE: The Commission meets this afternoon in one of, I gather, very long series of meetings which at some place I read goes back eleven years, addressing anticipated transients without scram.

In the recent, very recent past on this subject - that means within the last few months, the Commission did have a meeting with its staff to hear a briefing on the final staff proposal on the proposed rule. Prior to that, we had received several letters from the Atomic Industrial Forum and from General Electric, requesting the opportunity to participate in a meeting to present some views. I gather we also received a request from EPRI.

We responded to, on behalf of the Commission, the Secretary responded to AIF, to EPRI, and to General Electric saying that we did invite them to an open meeting regarding this subject. The meeting is today.

I gather from the agenda that I have we will be hearing first from the Atomic Industrial Forum and then second from EPRI. I will turn it over to you, Clark, and I guess you will introduce your colleagues and also, I guess, introduce yourself and mention why General Electric is not here.

PRESENTATION BY AIF
D. CLARK GIBBS, VICE PRESIDENT, MIDDLE SOUTH ENERGY INC.; DIRECTOR, NUCLEAR ACTIVITIES (MIDDLE SOUTH SERVICES); CHAIRMAN, AIF COMMITTEE ON REACTOR LICENSING AND SAFETY, (ACCOMPANIED BY D. A. R. BUHL, VICE PRESIDENT, TECHNOLOGY FOR ENERGY CORPORATION; G. C. SORENSON, CHAIRMAN, AIF CRLS, SUBCOMMITTEE ON ATWS, WASHINGTON PUBLIC POWER SUPPLY SYSTEM; F. STETSON, MANAGER, RL & SAFETY PROJECTS, AIF.)

MR. GIBBS: Gentlemen, it is an honor and pleasure for me to be here today. My name is Clark Gibbs. I am director of Nuclear Activities for Middle South Services and vice president of Middle South Energy, Inc., the owner of the Grand Gulf Nuclear Station. I am here today as chairman of the AIF Committee on Reactor Licensing and Safety. I am also a member of the AIF Policy Committee on Nuclear Regulation and the EEI Executive Advisory Committee on Nuclear Power.

The statement on ATWS that I shall make before you today has the endorsement of these AIF and EEI committees as well as the members of the American Public Power Association's Nuclear Power Task Force which currently own and operate nuclear power plants on their systems.

I will be reading my prepared presentation to you because of the organizations which I represent here and the need for their considered review of my remarks.

CHAIRMAN AHEARNE: That reminds me of many items of testimony I have given before the Congress, joined by my colleagues.

MR. GIBBS: If you have any questions during these prepared remarks, do not hesitate to interrupt. I am joined here today by Fred Stetson of the AIF staff on my far left. On my immediate left is Jerry Sorensen, who is chairman of the AIF ATWS Subcommittee, and Dr. Anthony Buhl, on my right, who is vice president of Technology for Energy Corporation, all of whom will assist in dealing with your questions. Also present are others from the industry whom I may call upon should the need arise.

Both the NRC and the industry are vitally interested in the safety of nuclear power, largely for the same reasons. Those of us who advocate continued and expanded use of nuclear power have grown accustomed to the attention to detail, energy, and commitment that the assurance of nuclear safety requires.

We well understand the potential consequences of errors in judgment on public acceptance, unit availability, and cost comparisons with alternatives. Those of us who are owners of these plants are keenly aware of the importance that our ratepayers who live in the environs of our plants, attach to nuclear safety. We have not failed to observe as well the hideous financial impact attendant with an event which compromises our ability to provide adequate cooling for the reactor core. We have every reason to be the most committed to nuclear safety of any organization participating in its use.

It is from that perspective which we view the ATWS issue, one which has confounded over ten years of attempted resolutions. We believe that the underlying reason for the inordinate length of time and effort that has already been expended on this subject, and which has frequently been spiced with ac -bic dialog is that it is an unprecedented attempt to pr.vide protection for a single extremely small probability event, from among a host of others which may have a greater probability of occurrence and for which the consequences are likely to be more severe.

We wish to enhance as necessary the safety and operability of our plants in a fashion which is self consistent, and objectively allocates our resources toward the achievement of a well-understood safety goal based upon a firm foundation of analysis of benefits and competing societal risks. In fact, it appears to us that the treatment of this subject by the NRC staff has been clearly overtaken by the events which have occurred since the accident at Three Mile Island.

The specific events to which I allude are the renewed interest in the establishment of quantitative safety goals, the ongoing and planned probabilistic assessment studies and the planned degraded core rulemaking. It is from these activities that we propose that the ultimate resolution of ATWS be derived.

In the interest of expanding upon this proposal, we suggest that the first prerequisite for a final ATWS resolution is the definition of a safety goal for nuclear plant regulation. The optimum ATWS resolution involves the reduction of risks that are already very small. Since it is impossible to reduce risks to zero, we continue to be confronted with the question, "How safe is safe enc ıgh?"

Although, of necessity, the lack of a safety goal has not precluded rulemaking in the past, it would be unwise to ignore safety goal guidance that should soon be available. Recent recognition that such guidance is essential suggests that it will be available in time to guide a final ATWS resolution.

CHAIRMAN AHEARNE: Clark, at this stage, are you speaking of a program we have under way that develops the safety goals, or are you speaking of something else?

MR. GIBBS: I am speaking of both that, and also I am hopeful that that process will enjoy the interaction with the industry in the evolution of the ultimate safety goals.

CHAIRMAN AHEARNE: Well, that process does involve interaction with all elements of the affected public, industry, etc. I wondered whether you had something separate in mind.

MR. GIBBS: No, sir; I do not believe I have anything separate in mind. I will say more to that as I proceed.

I should point out at this juncture that the AIF Committee on Reactor Licensing and Safety has recently come forward with a proposed safety goal before the ACRS which has received support within the industry.

A second prerequisite for a final ATWS resolution is further work on probabilistic risk assessment analysis. The last comprehensive PRA - which is the term I will use to refer to probabilistic risk assessment - which has been performed and widely circulated, and which treats ATWS among all the other events that can lead to degraded core cooling conditions was WASH-1400. That study suggested that the risk from ATWS events in LWR's was small. Other NRC studies such as the four volumes of NUREG 0460 have treated ATWS in greater detail than WASH-1400 but have done so in isolation or have compared a revised ATWS risk with unmodified WASH-140[illegible] [illegible]alues for competing risks.

This is clearly inappropriate and particularly so in view of the significant work underway and planned to expand the base of our knowledge in the area of PRA. Within the industry a growing number of PRA evaluations are scheduled for completion in the near future that will provide insights on ATWS.

The third prerequisite for final ATWS resolution is the integration of ATWS into the planned degraded core rulemaking. This rulemaking will determine whether and to what extent degraded core or core melt accidents must be considered in safety analyses. The end result of this process may be a rule that will amend 10 CFR 50 to require changes in plant design or procedures that will improve the capability of light water reactors to prevent, respond to, or accomodate the effects of accidents resulting in a degraded reactor core.

As noted above, the industry does not believe that final ATWS resolution can be achieved independent of the degraded core rulemaking. A systematic safety evaluation of a nuclear power plant should consider all the sequences and suggested modifications in perspective. In this manner we can direct our attention and resources to the dominant sequences that impact safety as well as to events that could result in other severe consequences.

COMMISSIONER GILINSKY: Did you really mean it when you said to interrupt you if we have questions?

MR. GIBBS: Yes, sir.

COMMISSIONER GILINSKY: What I am wondering is, what singles ATWS out here? It seems to me the things brought forward would apply to any number of other safety issues. It seems the suggestion that we ought not to move forward on these until we straighten out our philosophical framework, the safety goal, and a whole bunch of other things.

Is there something about ATWS that singles it out?

MR. GIBBS: No, sir. That is exactly the point. The risk associated with ATWS is one of degraded core. What we are suggesting here is that it be treated as such, along with the other scenarios which can lead to degraded core.

Because the same issues and facts are crucial to each, that is each of the potential degraded core scenarios, ATWS is simply a sub part of the degraded core matter; we recognize that the risk of ATWS, to the extent that there is any significant risk, is one of degraded core.

We recognize that ATWC is one relatively low-probability event among many that could conceivably lead to a degraded reactor core. Accordingly, there seems to be no sound reason for seeking final ATWS solutions for plants in isolation from other degraded core events.

We would prefer to avoid continued dialog on ATWS independently, and therefore propose the matter be disposed of now in a fashion which is supported by the record and which results in a substantial reduction of the ATWS risk. The stage has been set to treat the residual ATWS risk in the degraded core rulemaking in a fashion which will be acceptable to the industry and in particular to the owners of these plants.

There remains the question of what can and should be done now. The staff has recently proposed an ATWS rule and regulatory guide contained in SECY-80-409. You have also been served with a petition for rulemaking by the ATWS Utility Group representing 20 domestic electric utility companies. The two proposed rules are quite similar insofar as specific short term hardware requirements are concerned. Beyond that, they diverge. In the longer term, the staff proposes to specify criteria rather than mitigating hardware.

We believe this is a significant positive step and that a final rule which may evolve as a product of the degraded core rulemaking should rightfully address itself to criteria rather than hardware.

However, the proposed criteria are premature and as a result deficient. In our judgment, the staff proposals do not provide closure of the ATWS issue. The proposed regulatory guide will afford the staff unrestricted opportunities for imposing further regulatory requirements which will inevitably result in

ATWS becoming a design basis event for structures, systems, and components with implication far beyond that of which any of us today are capable of imagining. The appearance of a new design basis event virtually guarantees substantial impacts on the resources of both the NRC and industry for many years in the future.

The proposed integral plant and separate effects testing identified in the Regulatory Guide are briefly outlined as to purpose only. There is no way of intelligently evaluating what is expected of us from these purpose statements and certainly not in the time allowed for in the schedule which I will address later.

Further, the appearance of these tests is additional evidence that the staff is moving in the direction of treating ATWS as a design basis event after the fashion of the design basis loss of coolant accident, a practice which led to some of the unfavorable findings of those charged with the task of evaluating NRC's performance following the Three Mile Island accident.

The staff proposals are particularly deficient in the associated value-impact analyses, proposed schedule for implementation, and attention to detail where contradictions clearly exist in the record.

COMMISSIONER GILINSKY: Can I go back to the greater core rulemaking? There, it seems to me, the question is how much further should we go beyond the historic regulatory program in considering situations in which a core is in fact degraded and we might in fact want to take further steps to mitigate consequences, deal with hydrogen evolution, or whatever.

MR. GIBBS: Or to what extent preventive measures should be also incorporated or augmented to prevent degrading the core in the first place.

COMMISSIONER GILINSKY: I was going to get to that. It seems to me we have always tried to keep degraded cores from occurring in the first place. The element that the rulemaking would add - if we decide to make changes in our program - is a step beyond that envelope within which we have worked.

MR. GIBBS: Yes, sir.

COMMISSIONER GILINSKY: So, I guess I do not follow your logic in saying that we ought not to be trying to prevent cores from getting damaged, or take steps to prevent it, until we have been through that rulemaking. It seems to me that deals with questions that go beyond the ones we are talking about here in ATWS.

MR. GIBBS: No, sir, I don't believe so. As I conclude this statement, you will see that I am suggesting that we go ahead with certain measures which can offer preventive features with respect to degraded core matter. Our concerns are multi-faceted. Many of our concerns are that the fixes which may evolve as a result of the application of these criteria may result in fact in the reactor plant becoming less safe than it currently is; or in safety being degraded. We feel that these matters are sufficiently complex that they warrant further study.

Now, to the extent that there does exist risk associated with ATWS, that risk is all degraded core. Even after fixes are incorporated in these plans, there still will remain some residual risk which, we are suggesting, be bolted into the hopper of degraded core.

COMMISSIONER GILINSKY: When you say the risks come from the core being damaged, or degraded core, all risks connected with reactors come from the core being damaged and degraded, the integrity of the core not being maintained.

I guess I just don't follow your argument here. Am I missing something?

MR. GIBBS: That is a true statement, all risks ultimately arise in degraded cores, ultimately.

COMMISSIONER GILINSKY: But you seem to be saying that we ought not to do anything until we have been through this rulemaking.

MR. GIBBS: I am not saying that we ought not to do anything.

COMMISSIONER GILINSKY: The implication that I drew from this, that we ought not really to go forward not only in ATWS but on other fronts as well because the argument seems to apply there, too. I don't mean to derail you here from your presentation, but what I am grasping for is, what is is about ATWS that leads you to think that it ought to be handled differently than other parts of our concerns?

MR. GIBBS: I think Dr. Buhl should perhaps expand a little bit on this because apparently I am not communicating completely with you.

But ATWS has traditionally always been included, for example, in the WASH-1400 analyses as one of the events which can lead to a degraded core, which in turn can lead to risk to the public, both individual and population dose risk.

COMMISSIONER GILINSKY: Right.

MR. GIBBS: It is one of those events.

COMMISSIONER GILINSKY: Why don't you go on? I think it will sort itself out, maybe I am just missing something.

MR. GIBBS: Dr. Buhl?

DR. BUHL: Just to add a comment, I think ATWS is one of many sequences which, if you look at the dominant sequences in WASH-1400 for PWR, BWR. What we are saying is, there are certain things that one should do, and Dr. Gibbs will be proposing some of those in a few moments.

But our concern on the other hand is that if one looks at one accident sequence such as ATWS in the abstract, which is very easy to do, one might go too far, so to speak; that is, one might make a correction or at least a modification which he perceives to be a correction to reduce the ATWS risk and at the same time substantially increase the risk from these other accident sequences. So, I think the argument is that insofar as public risk is concerned, once you take a wholistic look and be very careful about dealing with ATWS or any other specific accident sequence, for that matter, in the abstract.

COMMISSIONER GILINSKY: Well, it is pretty hard to argue with the proposition that we ought to look at these things and be sure when we fix one thing we do not make some other things worse. But we were just discussing the action plan this morning, and there are any number of fixes that we are putting in place to deal with one or another sequence that we are concerned about.

It would seem to me that if there is a real problem here, we ought to be dealing with it. Of course, at the same time stepping back a bit, to make sure that we are not fouling up the rest of the system.

But you have tied your argument somehow to this degraded core rulemaking and I just don't see any particular connection.

MR. GIBBS: Sir, the connection is, I believe, that regardless what one does with his plant to deal with ATWS, there will remain some residual risk.

Some of these things he does can offer competition from other event sequences which can also lead to degrated core situations. The subject is very sophisticated and very detailed, and there is a great deal of system interrelationship involved which all deserve more attention than they have received. All of them, ultimately, lead to, when taking the worst path along the "event trees" degraded core conditions.

We believe that once the modifications have been made that we are suggesting here today in these plans, that it would be appropriate to deal with that residual in the degraded core rulemaking.

COMMISSIONER GILINSKY: Maybe it will become clearer as you go on.

MR. GIBBS: Beginning with value-impact, the Nuclear Regulatory Commission has adopted a policy, "That value-impact analysis will be conducted for any proposed regulatory actions that might impose a significant burden on the public (where the term public is defined in its broadest sense)." Consistent with this policy, the NRC staff has attempted to develop the required value-impact analysis for ATWS.

The staff's effort to date, however, has not been adequate. The major defects include first, failure to realistically consider the consumer impacts associated with major backfits and extended outages that will increase the cost of electric power.

The staff reports in SECY-80-409 that it is their judgment that extended downtime required to retrofit will likely be minimal. In view of the apparent need to provide additional relief valve capacity to meet the acceptance criteria of the proposed rule in Babcock & Wilcox and Combustion designed plants, this statement is profoundly in error.

A recent study performed at Duke Power Company indicates that a minimum of 31 days of additional down time would be required to make the pressurizer modification on Oconee necessary to provide the additional relief protection mandated by the acceptance criteria, assuming absolutely no problems - a most unlikely assumption.

The study further estimates that this unavailability would be likely to grow to 55 days if expected problems manifest themselves, such as difficulty in removing the pressurizer manway, or repair of indications on the nozzle welds. Approximately 360 man-rem of occupational exposure would be involved on each unit. Using $2,000 per occupational man-rem and $200,000 per day per unit cost of replacement power, which for Duke is nearly all coal, they estimate a $25 million impact on their three Oconee units exclusive of engineering and equipment costs.

It is important to point out that Oconee operation has been relatively free of fuel failures and their resultant exposures will be considerably below the average when plants which have experienced operation with failed fuel are taken into account.

In addition, the Oconee containment is relatively uncongested, minimizing the interference problem which will be experienced by others. Many other utilities will be required to use oil as a replacement fuel. It is therefore judged that the Duke estimates probably represent a lower bound on the cost of this single modification to the CE and B&W designed reactors.

Because of the sensitivity of this analysis to cost of replacement power and outage time, the ultimate relative impact to some utilities may be a factor of five or more greater than that suggested by the Duke study.

Another consideration is that the full implementation of the NRC-proposed resolution may also reduce system availability and reliability by making nuclear plants more complex and therefore more subject to malfunction when events such as inadvertent initiation of the automatic stand-by liquid control system are taken into account.

CHAIRMAN AHEARNE: was it that aspect you were referring to earlier when you mentioned it could reduce safety?

MR. GIBBS: No, sir. This event could contribute, but it is unlikely because there are other events. For example, it is being proposed that an automatic feed-water runback be incorporated on boiling water reactors. Inadvertent initiation of that feature challenges the RHR system. If that challenge occurs once per plant in its lifetime, the risk associated with failure of that RHR system, has been reported to me, is about equal to the decrease in risk that one achieves by full implementation of the ATWS fixes.

A second deficiency in the value-impact analysis is the failure to consider the increased risks from accidents other than ATWS that would be imposed by certain of the staff's alternatives.

Third, the value-impact information contained in

SECY-80-409 is nearly impossible to follow or understand. Discussions of value-impact estimates are contained in Enclosures B, F, and H of the document. These discussions are disjointed and confusing, referring to one or more different volumes of NUREG 0460 with various designations for the proposed fixes and contain unfounded and excessive dollar values for man-rem exposure. Further, the details of the modifications assumed as the basis for the impact estimates are not stated.

Fourth, the failure to recognize that few ATWS events have the potential of leading to severe ATWS consequences, that a limited set of severe ATWS events would result in major core degradation, and that not all major core degradations exceed 10 CFR 100 guidelines further results in the values being significantly overestimated and is not appropriate for value-impact analysis.

Turning now to the schedule proposed in SECY-80-409, it is safe to assert that it is unachievable and unjustified in view of the number of issues that remain open. We are being asked to submit evaluation models and plans for confirmatory testing by March 1, 1981, and to propose necessary modifications to meet the criteria by July 1, 1981.

It is clear that such a schedule allows no time to do anything other than fall back to the prescribed hardware "fixes" so much in evidence in NUREG-0460 Volume 4. If criteria similar to those presented in the proposed rule are ultimately determined to be necessary, substantially more time wil. be required to test alternative solutions, perform the detailed engineering, and perform the necessary reliability analyses to give us confidence that we are not "fixing" our plants in a fashion that will degrade rathern than enhance safety. Again, we need more experience with PRA methodoloty and implementation acquired on base studies before we begin to apply its results to making modifications to our plants.

The schedule further requires that boiling water reactor modifications required to meet the acceptance criteria be complete by July 1, 1982. On the basis of a proposal I have received from the affected vendor in this case, I know this to be unachievable.

We expect the same to apply to the PWRs. Finally, the significant pressure boundary work that may be required on the affected PWRs is to be complete by January 1, 1984. Should pressure boundary backfitting in fact be required, there is a time for doing that, and it is during the ten-year in-service inspection. Reserving any such modifications for that inspection availability will substantially reduce the impact to the ratepayer from nuclear plant down time.

Our problems with the achievability of the schedule are not limited to the plants which now have or expect operating licenses by January 1, 1984. For example, using the proposed schedule, the applicant for a nuclear unit expecting to receive an operating license in January, 1984, should have submitted proposals for complying with the recently announced criteria in

January, 1979.

We see no reason for including detailed implementation schedules in rules and suggest that such a practice not be continued here. The staff certainly has at their disposal alternatives to the establishment of such schedules short of including them in the rules.

Another major deficiency concerns the question of the staff's lack of attention to technical detail. A major portion of industry perceives the staff's "engineering judgment" in this area to be deficient. For example, the staff assumes that all ATWS events that could lead to a core melt will exceed 10 CFR 100 limits. These assumptions are overly conservative.

They ignore the fact that exceeding stress Level C requirements or exceeding an arbitrary temperature limit in a boiling water reactor torus does not necessarily lead to core melt, and core melt does not necessarily lead to violation of containment integrity or to exceeding the 10 CFR 100 limits.

They have not taken into account any operator action which, for such an event, would be a certainty. They overestimate the number of significant events because (A) below a certair power level, the consequences of an ATWS are not significant; (B) many anticipated transients when combined with a failure to scram do not lead to bounding consequences; (C) the consequences are a function of time in cycle; (D) not all ATWS events will necessarily cause a complete failure of the reactor shutdown system; (E) an ATWS event need not necessarily cause a failure of the reactor control system; and (F) as the experience level rises with added years of operation, the number of significant events falls for certain categories of initiating events as a result of the learning curve.

The staff has not treated in appropriate detail evidence that some of the measures that have been recommended to decrease the ATWS risk may, in fact, increase competing risks, thus lowering overall safety. The example that I cited is one.

CHAIRMAN AHEARNE: Evidence, I assume, carries with it some detailed analysis or actual case histories, as opposed to the hypothesis.

MR. GIBBS: Well, many of these analyses have been performed, and many of them have been submitted to the staff.

CHAIRMAN AHEARNE: That is what you meant by it.

MR. GIBBS: Yes. There is also an increasing data base of this evidence by virtue of studies that are currently under way in this area.

Approximately 20 utilities representing about 60 plants have proposed a solution recently in the form of a petition for rulemaking on ATWS. Part 1 of the petition proposes modifications that are straight-forward and well understood by the industry and the NRC staff. Thus, these modifications will not require great expenditures of resources for technical analysis, and they can be implemented quickly. Because a substantial portion of the industry is already willing to make these modifications if they will resolve the ATWS issue for existing plants, there is not likely to be much regulatory effort required to impose them. Most important of all, the proposed modifications clearly decrease the risk of ATWS while minimizing other, competing risks.

COMMISSIONER GILINSKY: Since you put the industry's willingness to make these modifications in terms of NRC's willingness to call it quits, does that mean that you do not really think even these are needed?

MR. GIBBS: I think that certainly there are elements within the industry who do not think these are needed or useful. I think the vast majority of the industry would be willing to go along with modifications such as those that I am about to propose if this dialog is brought to a close.

COMMISSIONER GILINSKY: It does not sound like something you want to do on your own.

MR. GIBBS: I do not view the prospect with a great deal of enthusiasm, no, sir.

In addition, the petition proposes that if the Commission elects to propose ATWS modifications beyond those in Part 1 of the petition, then all concerned will find themselves in a morass of unanswered questions demanding immediate answers and excessive NRC and industry manpower requirements.

Chief among these questions will be whether the additional potential modifications, if implemented, would leave the public more safe or less safe. The petitioners indicate that nothing short of an ATWS rulemaking involving adjudicatory procedures could provide the answer. The petitioners urge that such a rulemaking be held if ATWS modifications beyond those in Part 1 of the petition are, in fact, to be considered now.

We feel that such action coming at this time on this event would be unwise and counterproductive. Doing so would be an attempt to provide the ultimate resolution of ATWS in isolation from all other degraded core scenarios. One of the first lessons learned from Three Mile Island was that NRC and the industry had concentrated too much on low probability events. We should not forget this lesson in our efforts to improve the safety of our plants.

In conclusion, the organizations that I represent here today hereby recommend the following:

1. That the staff proposed acceptance criteria for analysis of ATWS mitigation capability although well intended, are premature and should not be adopted at this time.

2. That the Commission accept the utilities' proposal contained in Part 1 of the ATWS Utility Group petition. Doing so will reduce the risk associated with ATWS by at least 50 percent.

3. That a decision on whether additional risk reduction is appropriate await the establishment of a safety goal and the insights to be gained in the near future from the several on-going probabilistic risk assessment evaluations.

4. That as a result of the above, the unresolved safety issue on ATWS be closed now, and any residual risk be treated in the degraded core rulemaking.

CHAIRMAN AHEARNE: Thank you, Clark. Do any of your colleagues wish to add remarks?

Now, do you also represent General Electric?

MR. CHILK: It is my understanding that General Electric joined with AIF.

MR. GIBBS: General Electric is a member of AIS, sir. They had representatives on our committees, including my Steering Group on Reactor Licensing and Safety. They called me last week, as I recall, and indicated their intent not to participate, that they felt they were getting adequately represented by this paper.

CHAIRMAN AHEARNE: All right, thank you.

COMMISSIONER HENDRIE: Would you contrast the hardware changes that are proposed in the industry petition with the staff's what I call two-way, or basic short-term modifications?

MR. GIBBS: Yes, sir, if you will bear with me a moment.

Appendix D of SECY-80-409 contains a discussion of alternatives. The ATWS Utility Group proposal begins at page 6 of the petition for rulemaking.

In the case of the boiling water reactors, to begin with them, alternative 2(a) contained in SECY-80-409 contains an ATWS rod injection system which is also present in the petition. It contains a scram discharge volume modification. There are minor differences between the staff proposal and the industry proposal; but they are both treated.

It contains a recirculation pump trip which is also a utility petition.

There are two items which it also contains that are missing from the utility petition. One is logic changes to lower the low water level set point for initiation of containment isolation. That appears to me to be a minor matter.

COMMISSIONER HENDRIE: Low water set point where?

MR. GIBBS: On the reactor level. The reactor level trip set point for initiating containment isolation. The staff is proposing that that level be lowered in order to reduce MSIB closure ATWS type events. Now, the petition is silent on that. My understanding of that is that that is a relatively minor affair. However, it should be nonetheless looked at through a PRA-type analysis.

The only significant distinction between the staff proposal and the petition is, the petition does not contain any requirement for feedwater logic, feedwater runback. Now, that is the distinction.

COMMISSIONER HENDRIE: Between BWRs?

MR. GIBBS: For BWRs, yes, sir.

Now, in the case of the PWRs, first the Babcock & Wilcox and Combustion design plant. The staff proposal contains an AMSAC(?) which is present in the petition -- excuse me, the AMSAC which is defined in the petition is limited to an automatic initiation of auxiliary feedwater independent of the reactor protective s;stem, whereas the staff proposal is more general.

The staff proposes an alternate rod injection systen, as it is called, a supplementary protective system in the st f proposal, which is also present in the petition.

The staff also proposes analysis. The petition is silent on that.

With respect to the Westinghouse plants, the staff proposal includes a back-up scram system which is missing from the petition. The staff also includes an AMSAC, the petition only refers to automatic feedwater, auxiliary feedwater initiation.

So, those are distinctions insofar as the short-term requirements are concerned.

It is my belief that with the single exception of the feedwater runback on the boilers those distinctions are minor.

CHAIRMAN AHEARNE: Any questions?

COMMISSIONER HENDRIE: I guess not a question, more a comment. The reason this has gone on for so many years is that we rage back and forth over the argument as to what the probability of a serious ATWS in that might be, and whether that probability lies low enough so that one might be willing to regard it as an acceptable part of the inevitable residual risk, or whether it is high enough to require some specific design features, operating practices or what have you, to deal with it, either by way of prevention or by way of mitigation, or both.

We have hammered back and forth across that argument since 1969. It continues to be at the root of the disagreements over whether specific ATWS measures are required or not. I comment that I suspect the reason for having an enunciated quantitative safety goal seems such an attractive proposition to the other side of the table befc_e a final ATWS solution comes is just that it looks as though ATWS probabilities as evaluated by the assorted parties lie in the general neighborhood of where a quantitative safety goal might come out - probably with the uncertainty or the spread on those estimates running to either side of a reasonable safety goal.

So, depending on which side you are calculating from and so on, why, you either believe that whether or not you have an enunciated safety goal or not, you ought to do something about it. If you come at it from the other way, why, your belief might very well be that a reasonable safety goal, compared with ATWS probabilities, would show that nothing specific is required.

Since there has been a long history and a lot of people over various times have tried their hands on this, I doubt that these disagreements are apt to go away.

Well, let me stop there.

MR. GIBBS: Commissioner Hendrie, I wonder if I can make a comment.

CHAIRMAN AHEARNE: Sure.

MR. GIBBS: I don't believe, first, that one of the features of the safety goal which we expect to see emerge is individual risk criteria. My committee has suggested to the ACRS that the value that you can assign to that criteria should be tendered a minus five, and that is per year for the maximally exposed individual.

We do not believe that the individual risk from ATWS to the maximum exposed individual be anywhere near ten to the minus five. We believe it will be significantly less than that.

COMMISSIONER GILINSKY: Ten to the minus five what?

MR. GIBBS: Ten to the minus five per year for the maximally exposed individual.

COMMISSIONER HENDRIE: Ten to the minus five per reactor year chance of what serious radiation exposure, immediate, long term?

MR. GIBBS: Immediate. We fold the long term in with another part of the safety goal recommendation, which is the population dose criteria which is, I don't believe, at issue on this particular event, although it may well become an issue.

We don't believe that we are close to that criteria. In fact, we don't believe that, regardless of what criteria emerges, that when it is phrased in that context, i.e., the risk to the public, that the contribution from ATWS will be even close. We believe that other degraded core events in areas will dominate.

And the bottom line is, we don't know. We have to find that out. We do believe that the modifications that we are proposing here caused the risk to be sufficiently reduced in a WASH-1400 context that it is essentially in the background.

COMMISSIONER GILINSKY: How well do you think we know these risk probabilities?

MR. GIBBS: I don't think we know them well enough. I think that we have a decent understanding of them. But I don't know them well enough. I think that we will know them much, much better two or three years from now than we know them today.

COMMISSIONER GILINSKY: Well, that argues both ways, it seems to me. You may decide since you don't have a real firm grip on the numbers, you may just decide to follow a kind of common-sense approach and protect against certain contingencies whether the number is exactly right or not, simply because these are important possibilities.

MR. GIBBS: Yes, sir, I agree, that is an accurate statement.

The reason for my confidence, however, in response to Commissioner Hendrie's statement, is that the staff's estimates - of any group that I would expect to make conservative estimates with respect to these risks, I would expect them to come from the staff. The staff estimates that the risk from ATWS is something like eight times ten to minus five, as I recall. That is the number that they used to go into their value-impact analysis.

What I am proposing here will reduce that risk by a factor of at least two on the most affected plants, and that will be confined only to the plants of two vendors. Then, in addition to that, I pointed out there are a number of options available to the operator for further reducing that risk. There are a number of actions that are obvious, that he can take.

Then, on top of that, one has to concern himself not with what is the probability of this severe ATWS event having occurred, but what is the impact to that maximally exposed individual on the site boundary, and that is the parameter which appears to us to be of greatest interest. We believe that is substantially less, that there will be a very small number which will be multiplied by whatever the residual risk of a severe ATWS is, to calculate what the risk to that maximally exposed individual will be.

COMMISSIONER GILINSKY: What is your reaction to the Brown's Ferry the chairman mentioned earlier, it has not come up so far.

MR. GIBBS: Yes, sir, I think that is an event which deserves some mention here. First, there was no anticipated transient. Second, less than half of the control rods were affected. Third, those control rods which were affected went in part way; and fourth, operator action was successful in getting them in the remainder of the way.

There were a host of other options that the operator had at his disposal, that he could have used to further mitigate the consequences of that thing, that he was not obligated to use. He went to his first line of defense, which was the shut-down system, and ultimately was successful in getting the rods in.

Now, that event is part of the learning process. By going through that process we have eliminated one more source of common mode failure. I think we should fix those things. But I don't think that we should chastize ourselves or flagellate ourselves as a result of having experienced that incident. I think that it was a learning process, and it was far removed from the classical ATWS which is all rods stuck out at a hundred percent power following an anticipated transient, and no ability to get them in.

COMMISSIONER GILINSKY: But still, that is another one of those things that cuts both ways. Sure, we want to learn from it and it is inevitable that we will find things we have not expected and that is the way, in fact, that you improve the system.

But it is also true, so far as I can tell, this is a common mode failure that was not foreseen and was sort of surprising. In this area and other areas we would come across things like that which suggest that maybe we ought to be a little more cautious. That is really what is involved here.

MR. GIBBS: Commissioner Gilinsky, that is precisely what I am proposing, that we be more cautious because on the other side of the fence, if we are careless with respect to what we do to these plants, to address these very low risk accidents, we can wind up with a plant on which the safety has been diminished by virtue of those actions.

COMMISSIONER GILINSKY: Well, we certainly don't want that.

MR. GIBBS: No, sir. I certainly don't want one as a representative of an owner.

COMMISSIONER GILINSKY: Let me ask you something else, you mentioned the recirculating pump trip as something you would propose - I guess it is required now and needs to be completed.

MR. GIBBS: Yes.

COMMISSIONER GILINSKY: I don't know when the date is.

MR. GIBBS: By the end of the year, sir.

COMMISSIONER GILINSKY: By the end of the year. Why was this so long in coming, is this something we have known about for a long time and yet, I gather, it has only been recently that clients have effected these changes.

MR. GIBBS: Sir, I don't know that I can give you a great deal of history on that. Although, if I consider what my reaction might have been when first confronted with the idea, it would have occurred to me that, gee, is it really a good idea to turn off your cooling flow immediately following a transient, which is effectively what you are doing with the recirculating pump trip. That may have been the cause of it.

Jerry or Fred, can you offer anything further on that?

COMMISSIONER GILINSKY: My impression is, you don't have much time to work with in the event of an ATWS if the pumps are not turned off.

MR. GIBBS: That appears to be the case, yes, sir.

MR. SORENSON: I think there has been some good information provided in that regard by Commonwealth Edison in some of their presentations previously to the ACRS regarding ATWS. I don't recall the details, but I think that might be something worth bringing back out and providing the Commission.

CHAIRMAN AHEARNE: I guess our time is almost up in this section. I just want to ask you a question in the statement in here, since I gather you associate yourself with the remarks.

MR. GIBBS: Yes.

CHAIRMAN AHEARNE: On page 2, at the bottom, let me drop out a word which is a "may" which is a qualifier. This is now talking about ATWS. "From among a host of others which" - I am going to drop out the "may" - "which have a greater probability of occurrance and for which the consequences are likely to be more severe." Is the "may" there essential?

DR. BUHL: I think the "may" is there because if you look in Appendix 5 of WASH-1400 for the PWRs and the BWRs, you can find those other sequences and simply see the numbers. In fact, I have done that. When you look at those numbers there for the PWR, they are comparable for a dozen sequences, or so;

and for the BRW the heat removal sequence is also comparable. In my view, they are comparable. So, that is what we are saying.

CHAIRMAN AHEARNE: So, there are other sequences which have a comparable probability.

DR. BUHL: Yes, sir. The numbers are in those tables.

CHAIRMAN AHEARNE: Clark, do you have anything else you want to add?

MR. GIBBS: No, sir.

CHAIRMAN AHEARNE: Thank you.

MR. GIBBS: Thank you.

CHAIRMAN AHEARNE: The next presentation is from EPRI. Dr. Lellouche, the forum is yours, sir.

MR. LELLOUCHE: Thank you, sir.

PRESENTATION BY EPRI

G. S. LELLOUCHE (ACCOMPANIED BY DR. IAN WALL, EPRI.)

MR. LELLOUCHE: Good afternoon, gentlemen. We are pleased to accept the Commission's invitation to contribute to these important deliberations.

Before starting I would like to make clear that this presentation is based on portions of the research done by the Institute, by their personnel, and their contractors in the area of probabilistics. As such, it does not represent a formal EPRI position. The formal EPRI position on ATWS is contained in the comments to Volume 4 of NUREG 0460 sent to Mr. Tadani(?) some time ago. In light of the fact that EPRI did not receive a copy of the current SECY document until last week, and then only from a secondary source, if the Commission wishes a further formal response to the staff's review of our position we shall be pleased to supply such when we are requested to.

I would like to start by referring to a rather old letter, 21 August, 1978, to Dr. Kerr from Carl Bennett, who is the ACRS ATWS Subcommittee statistical consultant. He stated that there were no statistical problems with the methods used by EPRI to treat the historical data. He suggested a modification in the procedure to combine plant data and we agreed to use the suggested method. He stated that the disagreement between the staff numbers and EPRI's came from the differing input data.

The input data arise from the following considerations -

and you gentlemen will forgive me, this is going to be a technical presentation.

CHAIRMAN AHEARNE: We will manage, I think, to hold through it.

MR. LELLOUCHE: It is always more difficult when you can't talk in generalities.

The input data arise from the following considerations: How many reactor years of experience are there? How many tests of the electrical system are there during a year? Is the kahl event pertinent to a calculation of ATWS probabilistics?. How many transients per year are significant from an ATWS viewpoint? Is the effect of bypass capacity pertinent to the number of transients impacting on ATWS? Is the initial power level of the reactor pertinent?

A second question relates to determining a scram failure probability by using fault tree systems modeling - the so-called synthesis method.

A third question relates to competing risk and whether the suggested fixes actually reduce total risk.

The EPRI analysis of the historical data is found to be consistent with the fault tree analyses. The effects of rod and drive failures is found to be only a very small fraction of the probabilities found using the fault tree analyses, and this conclusion is consistent with the results obtained by Messrs. Vesely and Easterling using statistical models that are not based on fault trees. The uncertainties in the historical results at the 5th and 95th percentile levels are also found to be consistent with the fault tree results at those percentiles. The effect of the square root bounding method for rod and drive is also shown not to meaningfully impact on a fault tree analysis above the 50th percentile.

Our analysis of the effect of adding valves as a PWR fix has been previously addressed with the ASRS, and was shown to increase the total risk by increasing the small loca probability. We shall in great detail re-examine this work and address other reactors besides those in WASH-1400.

May I have the first figure, please?

This viewgraph shows the reactor years of experience. The NRR staff has stated in NUREG-0460 that there were 659 years. As of six months ago, EPRI stated there are 900 years. There are now 950 years of experience. The staff has not changed their mind as yet.

The testing rate. The NRR staff says there are 12 tests per year.

CHAIRMAN AHEARNE: Per plant?

MR. LELLOUCHE: Oh, yes, per plant; certainly. In all this, except where otherwise stated, I believe I will be only talking about per plant information, except where it is obvious.

The EPRI position on this is that there are approximately a hundred tests per year. Let me go into that. What I am talking about here are tests of the electrical system. I am not talking about tests of rod and scram drive per se, but only the electrical system.

What staff has been doing over the years, in fact from WASH 1270 on, is listing an analysis of the tests of the electrical system. They have stated a number of times in the dc·uments that they believe the rod and drive to be much more reliable than the electrical system.

The mechanical engineering staff of the regulatory branch has stated that they believe the rod and drives to be much more efficient than the staff believe.

From this point of view we have proposed to bound t·e testing rate and its impact by looking at the electrical system since the staff believes that is the worst portion.

CHAIRMAN AHEARNE: Now, I assume that what you are saying is that you and the staff, when you sit down and debate it, disagree on the definition of a test. Clearly, you do not differ on the definition of reactor year. So, I gather the difference in the first is merely the date at which you choose.

MR. LELLOUCHE: One might arg e that they have not changed their mind yet.

CHAIRMAN AHEARNE: Or that they have enough data past 1978. I assume the years and plants are not in debate.

MR. LELLOUCHE: Probably not.

EPRI has in the past stated that the number of BWR tests per year is about 200. For PWRs we use 12 per year and state it to be an absolute lower estimate because we felt it difficult to account for the split testing procedures used at PWRs. We state today that the number of PWRs is also about 200 and we can, if you wish, discuss the mathematics which lead us to believe that the split testing is in fact equivalent to a full single test.

CHAIRMAN AHEARNE: Would you say about 200 per year?

MR. LELLOUCHE: We will see exactly how those numbers arise.

For each transient of significance there will be a number of trip levels reached. May I have the next slide, please?

They generally will be neutron flux level, pressure vessel pressure, BWR water level for BWR. There will also be trip signals associated with specific transient, turbine trip, MSIV closure, loss of condenser vacuum, feedwater pump trip, loss of offsite power.

Westinghouse, in its publication has presented this table for the particular trips that are reached for these four transients which are very important from an ATWS point of view. In all cases you will notice there are three.

Some will say, aha, you don't get a turbine trip because there is no turbine trip for certain types of anticipated transients. But in both these cases you see that you have two others as well, and in our analysis we shall only use two as the number of trips that are achieved. May I have the next slide, please?

It appears we have lost the slide. For BWRs the trip levels that are reached for loss of condenser vacuum or a stop-valve flux and vessel pressure, or MSIV closure, flux, vessel pressure and stop valves, and the same for turbine trip or generator trip, for pressure-regulative failure only two are reached, flux and vessel pressure; for loss of feedwater flow some three are reached, low water level, isolation valves - I am sorry - for flux and vessel pressure.

There are in all cases except a very, very few, and those very few have extremely small frequencies of occurrance, a minimum of two trip levels that are reached.

There will be other trips as well associated with steam generators. Each of those trip levels has associated with it a number of independent electrical channels, usually four. These channels are tested once every four weeks, yielding for two trip levels at four channels each -- well, I have it for three trip levels at four channels each, 144 tests per year. So, that would be approximately 80 tests, 90 tests a year for two trip levels.

Some transients reach a full 200 tests of the relevant channels per year because they hit four trip levels. Some less, approximately 100. In PWRs all channels enter a four-fold redundant low voltage relay, each of which is tested every four weeks, yielding 48 tests per year.

Now, in this particular list for Westinghouse we see that there are four channels for each of these, except for the low reactor coolant flow which are three per loop, which means many more than four. They are tested each 28 days, averaging approximately six a week. Then the testing continues from the bistable to the actuator. There are six pairs of channels, each are tested every 28 days. Two breakers each are tested over 28 days. May I have the next slide, please?

The same thing is true for BWRs, only for BWRs we do not split the testing. For BWRs the test is a complete one, going from the sensor to the valve lifters, and they are done approximately five a week. Those question marks should be four, done approximately five a week.

Next slide, please, for B&W, I hope - we have them all backwards. For B&W the same thing is true, only here it goes from bistable into logic and the logic has trip relays, and then you have the trip breakers themselves. There are eight breakers and they go in a one out of two followed by two out of four. These are tested 40 times a month, for the logic 24 times a month for the logic trip relays; eight times a month for the breakers, and approximately four to six times a week for the channels.

Finally, for B&W the same thing is true here. We appear to have lost something, but continue on. What is the next one?

COMMISSIONER GILINSKY: What is it that you are calculating here?

MR. LELLOUCHE: I am trying to demonstrate the number of tests of the portions of the system that are actually done each month. The purpose of this - these are all electrical, remember. The purpose of this is to demonstrate that in fact the electrical portion of the system from beginning to end, whether it is done split into two or three parts, or for BWRs as a single entity, are tested a number of times per month perhaps five, ten, 50, depending on which portions you are talking about - not once. Not once.

Each channel is tested once a month. There are many channels. Each breaker is tested once a month. There are for CE eight breakers. Trip relays are tested. Everything is tested many more times than once a month. When you add them together in a mathematically consistent way using statistical methods, statistical calculus methods, you find that in fact you have something like a hundred to two-hundred tests a year of the system, depending upon how many trip level sensors - whether you hit just a high pressure, or whether you hit a high flux as well, how many of those you hit. Per transient type you get anywhere from a hundred equivalent full tests of the electrical system to two-hundred equivalent full tests of the electrical system per year, not twelve.

COMMISSIONER GILINSKY: What were you saying, equivalent full tests?

MR. LELLOUCHE: Equivalent in the sense that for PWRs most, or at least some plants do not test them from beginning to end in a single unit, but they split testing from the beginning to the middle, and the next week the middle to the end. These are done for presumably good and sufficient maintenance reasons, but mathematically one can show that they combine when the frequencies are a failure or small - as they are here in fact - that this mathematical analysis show that they combine and it is fully equivalent, mathematically, to a full test.

So, I use the word "equivalent", so as not to be caught in a mathematical misstatement. They are mathematically equivalent to a full test of the electrical system, and there are a minimum of approximately 100 a year, not twelve.

That is all I am trying to do by demonstrating all of these multi-channel tests which go on.

The staff presumes in fact - I presume they presume - that all of these mean one test, one test of the total system, all channels of all sensors, all breakers, all actuators, everything. That, to them I presume, means a test. But from the point of view of what the reactor sees during a transient, that is n[illegible]. It is not true engineering-wise; it is not true mathematically; it is not true physically, it is simply incorrect to make such a statement.

Now, if we go on with this, 900 reactor years, a hundred tests a year, approximately 90,000 electrical tests of the system. May I have the next slide, please? The one that says, "Summary of Testing Rates" at the top. Please, remove the others that you have already shown. I am sorry, gentleman.

Now, this shows the testing rate of the various portions of the system. BWR is depending upon transients 100 to 200 times a year. Again, these are only electrical tests, tests of the electrical portions. PWRs, sensors to bistable 100 to 200 times a year; bistable to actuator, depending upon reactor; breakers themselves, depending upon reactor. May I have the next slide, please?

Now, we have approximately 90,000 tests. As I said, 100 tests per year, 900 reactor years, 90,000 electrical tests. If we apply the statistical methods used by the staff in fact, these are the staff's statistical methods, pi square, and neglect kahl, we get the top line.

With a median estimation of failure of the electrical portion of the system of approximately four times tenth of a minus six per demand. If we include kahl, we get approximately two and-a-half times larger.

Now, if we do move on fr m here, which is purely historical data, the actual number of reactor years, the actual testing procedures used in real plants, and go on to fault tree analysis, these fault trees come from WASH-1400. We did some updating of the data and definitions, WASH-1400 assumes three rods have to fail in a BWR; it is really five rods minimum in a close connection. They assume any three rods for PWR, there is a minimum of 30 rods, things of that nature. If you correct those, there are some physics, then for fault tree analysis you find the second set set of lines which again show that the median estimation is in line with or without kahl estimation. More so than it is with kahl, but I would not really care whether you multiply them by two, it does not make a significant difference.

Mr. Lewis of the Lewis Report has suggested that using square root averaging procedure is incorrect. He suggested that you should use the upper bound on multiple rod failure; and the upper bound is one percent - and he accepted this as not being unreasonable, one percent of the single rod failure. That is to say, if you have a hundred single rods, every time you have a hundred single rods failure, you have one total rod systems failure. We have not had a hundred single rods fail.

If we use that upper bound effect of the rods and drives, we get the pair of lines. Now, you will notice that it does impact significantly at the low end, at the five percent level. But it does not impact meaningfully above the 50-percent level, which indicates that the argument that the square root bounding technique is going to make significant changes. It certainly does not hold up under numerical analysis.

NRC calculated rod and drive effects,Messrs.Vesely and Easterling did them two different ways, not fault tree method, but they were using standard statistical failure models. They calculate failure rates significantly lower, in fact, at 99-percent statistical confidence level.

These types of comparisons lead us to believe that we are calculating, treating the data correctly. Our fault trees are correct, that is to say, consistent with data analysis; effects of things like upper bound techniques or square root models don't alter these conclusions, in fact. May I see what the next slide looks like?

COMMISSIONER GILINSKY: What is your summary of that?

MR. LELLOUCHE: My summary of that is that the failure of the system, the electrical portion of the system in a total failure mode, that is all rods failing out, would be in the neighborhood of three to five times ten to the minus six per demand; not what the staff originally calculated which was approximately --

COMMISSIONER GILINSKY: That comes more or less from the middle column.

MR. LEELOUCHE: That is the middle column, yes.

COMMISSIONER GILINSKY: Why do you take the middle column?

MR. LELLOUCHE: In a normal distribution the middle column would be the mean value. In "skewed" distributions to talk about the mean does not necessarily have meaning - if you forgive my pun. You do not know where it lies on distribution.

The mean, in fact, in this case would be something like a factor of two higher than the median. These distributions tend to be log normal not exactly, they have an early peak and a long tail; and it is hard to make a choice as to what kind of a confidence level to use. Why should be pick 95 over 99, why not five nines? What is wrong with the two-percent level?

The answer to that is, there is nothing wrong with any of them, it is just what you want to interpret. By choosing a median estimate we are erring on the side of equal error. That is to say, it is equally likely that we could be above or below.

If we choose a high estimate, the odds are very good that we are well below it. That might be considered conservative, but it could be considered too conservative. Where does one stop with conservatism? So, one might say the 99 percent, another might say the five nines level. It simply errs equally on either side. That is the best I can do. In a normal distribution it is the mean value.

Now, this is insufficient to determine ATWS. For ATWS we also need to know what the frequency of transients is going to be, and we take our list of one of the important transients from the staff.

The staff says that for B&W we have loss of offsite power; total loss of feedwater, and transients leading to loss of feedwater.

For combustion we have better discrimination.

COMMISSIONER GILINSKY: Let me return to that point. We are dealing with very small samples,failure samples.

MR. LELLOUCHE: Yes, correct.

COMMISSIONER GILINSKY: Therefore, your estimate is intrinsically an uncertain one.

MR. LELLOUCHE: Certainly, in terms of the failures themselves because we had so few. But not in terms of the tests, where we had 90,000.

COMMISSIONER GILINSKY: So, it seems to me in these circumstances, I guess, I want to think more about the 50 percent. It is sort of like reaching into an iron and taking out 10,000 balls and finding one red one.

MR. LELLOUCHE: That's correct.

COMMISSIONER GILINSKY: What do you conclude from that?

MR. LELLOUCHE: That the odds are something like one in 10,000?

COMMISSIONER GILINSKY: Maybe.

MR. LELLOUCHE: It depends on how many balls there are in the first place.

MR. LELLOUCHE: For combustion we have somewhat more detailed information for two different types of cores. The results for Westinghouse of consequence calculations, that is to say the transient that the plant undergoes shows that none of these transients, no transients exceed 3,100 psi at, I think, 95 per-centile moderated temperature coefficient. But the ones that yield the worst result - that would be something like 2,800, I believe - are loss of load and total loss of feedwater.

For General Electric, any transient leading to excessive cool temperature, is a transient of significance. May I have the next slide, please?

We collected from the utilities that would report them to us information on all their scrams, the origin of their scrams, the status of the reactor before and after the scrams, and we categorized these and published it as an EPRI document. I believe the staff makes use of it.

We took the staff's definition of what is a transient of significance, ATWS, and we broke them out from the EPRI analysis of real plant data. We found that there were these many transients for PWRs and these for BWRs. May I have the next slide, please?

If we quantify them, using the data that is in NP-801 which lists the actual frequencies, and we quantify them for the particular plants of necessity, LOOP is loss of outside power, loss of feedwater, loss of load; CEA is an uncontrolled rod withdrawal, etc., if we quantify them we find these numbers as being the numbers of events per year that occur.

CHAIRMAN AHEARNE: Now, you mentioned this was for the plants that would give you the data. Is that a large set of plants?

MR. LELLOUCHE: At that time it was approximately 50 percent of the plants and 50 percent of the reactor years. We have a new analysis, we have done a second collection because that one stopped in 1976. We have done a new collection. We now have 60 percent of the plant years and approximately 55 percent of the number of plants.

CHAIRMAN AHEARNE: Is this data based on that later collection?

MR. LELLOUCHE: The later collection has not been completely analyzed yet. We are still hoping to get some more data.

I can say this: These numbers do not change by more than three to five percent as far as my understanding of the numbers at the present time.

The staff, however, quotes different numbers. They also say that we have excluded a whole bunch of transients from our list. May I have the next one, please?

These are the transients that they say we have excluded. Of all of these, the vendors say the "nones" mean there is no effect, there is no significance. There is one error here on the BWR, the first one, on "stuck valves" the no should be a yes. I am not even sure you have it on your graph. It should be yes - yes - no - no of PWR.

They do not have much significance. The "maybe" there means those feedwater instabilities have to do with single loops. The scrams are mostly RX scrams.

CHAIRMAN AHEARNE: In your middle column there the frequency is at 25 percent?

MR. LELLOUCHE: Above 25 percent, I beg your pardon. I do not have data on stuck valves above 25 percent power.

The "maybes" refer to the fact that most of the feed-water flow instabilities, a number of them are caused by operators, about half; about half are mechanical, and almost all of them do not require a scram at the time it occurs, but a scram is an RX scram. Whether it would add in later on and cause trouble, I could not say. That is the largest ~~perri~~ error.

Now, if we go further, we now compare - may I have the next slide, please? We can take a look at what the power distributions are. The number of transients carrying below 25 percent power is approximately half, the total number of transients occuring for PWRs and approximately 70 percent for BWRs. The number of transients of importance to ATWS are also approximately half from between above 25 power and below 25 per-cent power. The importance of this will come up in just a moment. May I have the next slide, please?

If we plot up our numbers, the EPRI numbers, of the actual anticipated transients which are of significance and compare them with NUREG-0460, our estimate, we see that actual operating experience lies significantly below the staff's estimate - very significantly below the staff's estimate.

Now, the staff says that they do not believe that 25 percent power is a cutoff. May I have the next slide, please?

This is a CE machine, these are re Trend calculations done as a function of moderator temperature coefficient at power level with and without aux speed for two types of transients, lose of feedwater and MSIV closure.

Now, you will notice that for below 50 percent power, or certainly below 25 percent power, independently of the moderator temperature coefficient you don't exceed 2,500 psi, and that is true whether you have access to a condenser or not. That is to say, it is true for a loss of feedwater with condenser available and it is true for an MSIV closure without condenser and without aux ~~sp~~feed. The result is that calculations like this show that the staff's presumption that 25 percent power is an inappropriate cutoff simply is incorrect. May I have the next slide, please?

The next slide shows a B&W machine, this is for a loss of feedwater transient, and one sees here that the probability is not exceeding - this is unfortunately the negative of what I wanted to say.

CHAIRMAN AHEARNE: We have the one.

MR. LELLOUCHE: You have the good one? All right, you have the inverse of the slide. The probability of exceeding 3,200 reaches essentially zero below 75 percent power; 25 percent power is the cutoff. It is really quite an acceptable number for PWRs.

Now, for BWRs 25 percent power has to be combined with access to the condenser. You have a 25-percent or greater condenser bypass capacity, transients below 25 percent power or below the condenser capacity will not impact. Such calculations have been done, and I believe presented to the staff from Yankee, and they show simply that you don't get into any kind of trouble if your initial power level is less than your bypass capacity. Even when the bypass is not available, and you are going into the torus, if you are below 25 percent power you have approximately half an hour before you reach about 180 degrees, which is still 20 degrees below any staff limit. That is without turning on the torus, a heat-exchanger cooler. You have nearly an hour if you turn that up.

So, I would suggest that 25 percent power is a reasonable cutoff for all ATWS transients on PWRs and for all ATWS transients that have access to the condenser for BWRs. May I have the next slide, please?

The result of this is the frequency of transients and the ATWS frequency. We will look at the bottom of the table. The frequency of transients that are applicable to PWRs, if we deal with all PWRs as a unit, not separating them out by plant, is .6 per year. For BWRs approximately three and-a-half, and if we sum them up with a six to four split it is approximately 1.7 per year. These yield ATWS frequencies between two and two times ten to the minus five for Bs and three times ten to the minus six

Ps. May I have the next slide, please?

The staff, however, has used various numbers. In 0460 Vol 1-3 they have two times ten to the minus four; in Vol 4 they used eight times ten to the minus five; and for some PWRs, Westinghouse ten to the minus six, but they still used two times ten to the minus four for BWRs, and I would suggest that the real data yield numbers approximately in order of magnitude less.

Any safety-oriented plant modification contains within it the probability of accomplishing the goal desired, and the potential for creating new and altered pathways for accidents. Thus, the usefulness of any modification lies in a trade-off between the decreased risk inherent in the modification and the increased risk due to the new accident pathways created by the modification.

Examples of this trade-off are well known. Some of them are the interfacing LOCA (Event V of WASH 1400) where locking open an MOV to eliminate a single failure point for use of the LPSI increased the probability of the LOCA through the two check valves by a factor of ten.

Another example is requiring the auxfeed to actuate as a post TMI requirement for certain events has increased the number of pressurizer emptying transients which appear to the operator as a LOCA and increase the likelihood of operator misaction.

Closure of the blocking valves on the PORV and maintenance of the HPI has increased the number of safety valve actuations and in fact, it led to the safety valve actuation at Crystal River.

Each of these are competing ..k situations where unexpected results and increased risk are obtained from a supposedly safety-based modification intended to reduce it.

In the case of ATWS the staff has suggested that increasing the number of valves on combustion and B&W plants will reduce ATWS risk. The following analysis shows that this modification induces a competing risk situation and the increased competing risk is greater than the ATWS risk reduction. The competing risk here is a failure of a valve to reseat after it has opened, that is to say, TMI 2 and Cyrstal River.

In the following analysis we will consider WASH-1400 for a category characterization of the event sequence, but it will be made reasonable that for B&W and C.E. there should be no real difference. We shall also consider the Crystal River probabilistic risk assessment document and show that indeed for Crystal River ; this is also specific.

Now, ATWS risk. An ATWS event sequence - can I have the next slide?

COMMISSIONER GILINSKY: May I ask you, are those numbers comparable?

MR. LELLOUCHE: These numbers are all for comparable confidence levels; they are all median numbers. The staff numbers are median numbers and so are ours.

The next slide, please. An ATWS event sequence leading to potential damage depends on the time into the fuel cycle. Early in the cycle, insufficient fission products have built up so that a large amount of boron is in solution. Up to some time, T-1 say, in this figure, even if all the valves open during an ATWS, the moderator coefficient will be insufficiently negative to terminate the transient before an excessive pressure level is reached. In this time period the ATWS transient of importance is TK. That is to say a transient followed by a simple failure of the scram system. O-t stands for the frequency of the scram system.

In the second part of the fuel cycle the moderator coefficient is sufficiently negative so that if all the valves open, no excessive pressure will be reached. But if one valve fails to open - symbolized by P here - then an excessive pressure will be reached. Further on, two valves will have to fail to open. Further on, beyond that, three valves would have to fail to open. Beyond the point T-2 the moderator coefficient is so negative, even if all the valves fail to open no excessive pressure will be reached.

We estimate T-1 to be approximately 40 percent of the weight into the transient.

The only competing risk we deal with here is failure of a valve to reseat.

COMMISSIONER HENDRIE: How far out do you have to get on these plants to get T-2?

MR. LELLOUCHE: Eighty to 90 percent for some; it depends on the particular transient. If you have a 3,300 psi transient, not too much time; if you have a 5,000 psi transient you may never get there. But I will not be making use of T-2.

The only competing risk I want to deal with is failure of a valve to reseat. This event is denoted universally by Q. Clearly, for Q to occur the valve must have opened. The number of stuck open PWR valves has been determined by searching the LERs to be 9. May I have the next slide, please?

Using a 300 PWR reactor-year experience base, this leads to a transient frequency of stuck open valves of 03 per year. There are two types of sequences where failure to reseat is significant. The first is the ATWS event itself where the sequence - TKQ - leads to a small LOCA and any additional serious failure of HP.I leads to core melt. In WASH-1400 the additional failures will come up in a moment.

We have here besides these TKQXs, such as failure of the HP.I or failure of ECC injection, we also have the same type of event, that is to say a stuck open valve leads to release of liquid on the T*Q event. These are equally likely during the entire fuel cycle and don't have anything to do with moderator coefficient. Now, may I have the next slide, please?

These are the list of events that have occurred, we notice one of them is a blocking valve. In ~~Fort~~ Saint Calhoun we had a common mode failure of 2 PORVs, one was a safety valve and and a PORV valve. These two events have actually occurred. May I have the next slide, please?

The types of secondary failures chat have to occur are given their symbolism from WASH-1400. They are failure of containment spray injection; failure of ECC, the other three. Now, in the range of time zero to T-1 into the fuel cycle, TK is much greater than TKQX. In the second range, TKP is less, much less than TKQX, and TK onward, TKQX is the total range.

The types of transients which in fact lead to the lifting of valves are common to all PWRs and they are standard ATWS types when the scram system does occur, in fact. Their cold pressurization is one at which PORVs often open at a lower pressure level, but they have the likelihood of not closing again.

Now, if we go to WASH 1400 we can determine what these transient frequencies are in the sense that WASH 1400 deals with a small, small measure. They say that S-2 is like a stuck-open valve, and S-2-G which would be a small break followed by ECC failure at a certain frequency, and they listed the frequencies as they are listed on this graph.

When we take the S-2 frequency out, we determine what the actual failure rate of the secondary systems are, and from these we can now determine by summing them what the actual T*QX frequency is; and it is approximate y five times ten to the minus four.

Now, staff has said that their frequency is two times ten to the minus four, or eight times ten to the minus five, and if you take that 40 percent of the time - well, 40 percent of the time is the eight times ten to the minus five. If you calculate that as one point six times ten to the minus four and account for it, a valve lifting stuck open is approximately ten times larger than ATWS already.

Now, some reactors do not have the same type of failure modes as others. That is to say, Surry does not contain fan coolers for plants with fan coolers as well as sprays the C event would not be important. Similarly, plant variations imply that F&H are couples and that you should not differentiate, necessarily, between them. May I have the next slide, please?

We can recalculate T*QX for non-Surry types of plants to be approximately five times ten to the minus four, and ATWS is still approximately a factor of ten smaller than these stuck-open valve events.

Now, we have only looked at the melt probabilities. From a risk viewpoint this is insufficient, we have to look at the release probability as well, and the releast fractions. If I may have the next slide, please, we can see that all size scram failure and competing risk events are classified in the release categories 3, 5 and 7. The release magnitudes for 3 are much greater than for 5 and 7, and here we have compared them in terms of equivalent iodine which is a convenience only.

NUREG 0460 states that there is some possibility for an ATWS to be in PWR-3 but does not quantify this statement. It concludes the most likely failure mode is PWR-7. We quantify the probability of it being in PWR-5, in this case, by looking at TK-beta which is a category 5 event, and we include the TK-epsilon, which is a category 7 event, and we do not have to deal with any other categories as far as we know.

The competing risks are in two types, a delta risk and an alpha risk, the alpha risk being a steam explosion and delta being another form of core melt. If you take the ratio of these two - may I have the next slide, please - we determine what the actual competing risk is, and the competing risk is 5,000 times larger than the ATWS rate.

Now, that is true for Surry. If we neglect C&F modes of failure it is also equally true for non-Surry types of plants, and that is 500 times. If we go over to Crystal River - may I have the next slide. You do not have a copy of this, I made it up on the way in.

If we use the Crystal River report, T*Q us considered as a B-4 -- type event. The B-4 sequences are listed here, their melt probabilities, given a B-4 event occurs are listed here. Their sum is much greater than for Surry, the total T*QX is twice as large in Surry and the ratio of non-ATWS risk to ATWS risk for Crystal River is a factor of 10,000.

Now I don't care how many valves are put on, any additional valves will increase the risk. There is no conceivable way that additional valves can reduce the risk. These numbers are so large that even if only one percent of them went to increased risk, you would still have excessive increases in risk over ATWS.

I would like to very briefly now go through the comments made by the staff concerning the formal EPRI review of Volume 4. This makes reference to SECY-80-409, Enclosure F.

On page F2 the staff comments: EPRI should not use "much criticized square root bounding method," and that it improperly treated the Naval data.

We would respond to that by saying the EPRI analysis was in accordance with WASH 1400 to which it was being compared. To alter one analysis would have required redoing WASH 1400. Since only comparisons were being made no dichotomy exist.. Much more important, however, is the fact that the EPRI analyses of the historical data and the synthesis results utilizing the square root bounding method are numerically in accord with each other above the 50th percentile.

Further, the exclusion of the square root method does not alter this result. This result does not, however, depend on the use of Naval data. It does depend on estimating the testing rate in accordance, however, with actual plant practice and not with the staff's assumption.

Finally, the Vesely and Easterling NRC analyses predict still smaller system failure rates than the EPRI analyses did.

On page F3 the staff comment: The EPRI list of transients is incomplete.

Our response would be, if we add these extra transients in they make no significant difference in the numerics. In fact, however, the vendors dispute the additions of most of these transients. They do not agree that most of them would lead to trouble even if the scram system failed. In fact, some of these extra transients are already analyzed in vendor submittals and have been shown not to lead to trouble.

Second, the staff says that exclusion of events below 25 percent power may be inappropriate.

Response: Extensive evidence now exists that with the one possible exception of uncontrolled rod withdrawal which has a frequency of one per hundred years, the worst PWR transients which have access to the condenser do not lead to excess pressures when they start from below 50 percent power; and that this is true with or without aux feed. Both B&W and CE machines have been analyzed and the B&W analysis showed that this is true below 85 percent initial power; and the CE certainly below 50 percent initial power.

Comment by the staff: Only five years of EPRI collected transient frequency esperience data is meaningful.

We would respond: The staff has been saying this for the last two years. They should at least go to seven years. But the fact is that the PWRs, we have ten plants with nine or more years data and four with 13 or more years data. There is little rationale for excluding any data except for mathematically justifiable reasons, and no mathematically justifiable reasons have been made by the staff.

The staff comments: The EPRI analysis of testing rates is wrong.

We would respond: Channel tests are perfectly appropriate since they give upper bounds on multi-channel outages. Further, although the data indicates occasions where all channels of a given sensor type have failed, there is to our knowledge no occurrence of simultaneous failure of all channels of two or more sensor types, and almost all channels trip two or more types of sensors.

This is where the Fessenheim analysis which the staff uses as backup to their numbers falls apart. Although the data used to derive input for Fessenheim shows no simultaneous failures of all channels of two sensor types, the final result is based on a common mode failure factor of beta equal to .1. If this very true, there would with very high probability have been at least one and probably two simultaneous total failures of all channels of two diverse sensor systems.

Since such has not occurred it is more likely that beta equal .01 more correctly describes the simulteneous failure of two diverse sensor systems. This would produce results fully in line with the EPRI calculations of historical data and the

EPRI fault tree and other analyses.

More to the point, however, is the fact that the Fessenheim analysis has two arbitrary constants and their calculations may be made to any numbers desired. Further, the paper by Apostolakis which is also used by the staff as confirmation of their results has been completely discredited by commentators, including Mr. Easterling; and Mr. Apostolakis has agreed with the fact that his paper is incorrect. There is in fact no defensible calculation which backs up the staff results.

Page F7. The staff comments: Only stuck-open safety valves should be included in calculating increased risk due to staff imposed fixes.

COMMISSIONER GILINSKY: Mr. Apostolakis has agreed that his paper is incorrect, presumably he has done so in writing?

MR. LELLOUCHE: Yes.

COMMISSIONER GILINSKY: Where?

MR. LELLOUCHE: In his response to the letter which Mr. Easterling sent in as a letter to the editor.

CHAIRMAN AHEARNE: Has it been published in Nuclear Safety yet?

MR. LELLOUCHE: I don't know whether it has as yet been published in Nuclear Safety. I received it in the mail, and since I received it, they must have received it from Nuclear Safety. I am sure Nuclear Safety has it; and Mr. Apostolakis agrees that his calculation is wrong. If it in fact had been correctly done according tc his lichts, if he had done the numerics correctly, he would have gotten an order of magnitude higher than the staff's numbers. The staff's numbe.s would now be in the neighborhood of ten to the minus. His numbers, multiplied by the frequency of transient would be in the neighborhood of two times ten to the minus three per year. His analysis is incorrect.

It is incorrect for a n of reasons. If you wish I can go into them, but I don't know if you wish.

COMMISSIONER BRADFORD: It would be easier if you just sent us a copy of the letter.

MR. LELLOUCHE: I have to find it. I will get you a copy of the letter.

COMMISSIONER BRADFORD: Fine.

MR. LELLOUCHE: Now, on page F7 the staff says: Only safety valves should be included in determining increased risk from competing risks.

In response we would say: The staff thus throws out all LERs as being involved. They have rectified nine out of nine events. But on what basis? We agree that any additional valves would have higher set points, we stated so in just those words in our comments on Volume 4 of NUREG-0460. But firstly, most of the stuck-valve events were caused by human maintenance errors which had nothing to do with set points; and second, the new requirements concerning closing backup valves introduce additional failure modes, leading to safety valve actuation. We also stated this in our comments to Volume 4. It would seem that rectifying nine out of nine events would be slightly unconservative when the staff still refuses to agree on the rectification of the single kahl event because that would be unconservative.

On page F8: Operators have time to mitigate small LOCAS. The staff says that because of that none of these small LOCA events should lead to any problems.

We would respond that the staff might allow such consideration for ATWS. Browns Ferry clearly shows that the operator would respond rapidly to a failed scram. To assume that such operators would sit on their hands for ten minutes is clearly unrealistic.

On page F9 the staff says: The EPRI analysis is wrong because of errors and deficiencies.

We would respond, we put together the comments to Volume 4 in three or four days. There are indeed some typos and one numerical error. But the implication of the results still stands. The staff did not bother to correct the errors and requantify. Had they done so, they would have found that for non-Surry plants the valve risk is 500 times larger than ATWS risk. They did just not bother to do that.

Now, that is all I have to say, but we also present in our comments to Volume 4 an analysis of the value impact statement made by staff, and Dr. Wall would like to comment on that, if he may.

CHAIRMAN AHEARNE: Dr. Wall?

DR. WALL: I am sorry I do not have a whole statement since we only received the SECY document last week and I was out of the country.

We are very pleased indeed that we found our comment was acceptable, namely that recommending to use incremental rather than total values and impacts, the value impact statement.

However, it is rather unclear in the final SECY document, Enclosure B, where the latest table and value impacts states incremental and total numbers. I think Mr. Gibbs referred to that in his presentation.

We also found some inconsistencies in NUREG 0460 and recommended that NRC publish all calculation details. To our knowledge, such details are still not available, and indeed within this briefing table, Table 1, Enclosure B, is at least partially inconsistent with Table 3 where they are trying to correct the valves at page F11.

These frequent updates without full support is what renders it so difficult for other parties to track NRC's calculations. I think it would be very easy for the staff to help in that respect.

The NRC staff claims that EPRI misinterpreted their impact, Alternatives 2(a) and 3(a). EPRI as an R&D organization has no input to offer on impact estimates. So, we merely try to use NRC numbers. NUREG-0460 is somewhat confusing on the subject, so we rather carefully noted that the impact numbers were approximate in our comments.

In any event, the thrusts of our comments were on the estimation of value, not impact, and any misinterpretation only reflects on the relative impact of Alternative 2(a) and 3(a).

The staff's response is incorrect and in that respect seems overly encompassing and unjustified.

The NRC staff did not address Enclosure F, the other EPRI comments on their value analysis. These comments point out several deficiencies, some of which Dr. Gibbs referred to; for example, this variation in consequence magnitude for potential core melting accidents should be considered. When one does that, the rate of logical risk reduction claimed for PWR ATWS fixes are overstated by at least a factor of ten.

Realistic as opposed to conservative parameters should be used or, alternatively one wishes to be more informative, quote the realistic values and put a range which includes conservative values which may be more palatable to the staff.

The effect of proposed fixes on overall plant risk should be assessed as a whole and not just in isolation; and Dr. Gibbs addressed that very succinctly.

In summary, the NRC staff has not subsantitively addressed in Enclosure F EPRI's comments on the impact analysis. Insufficient reasons are given to change our contention that the values on ATWS fixes are greatly overstated. We believe that on-going policy risk assessments for specific plants will show that with a few "outliers" ATWS is not a dominant contributor to public risk. Thus, it will be more cost effective to address other --

CHAIRMAN AHEARNE: When you say a few outliers, do you mean plants?

DR. WALL: I mean maybe a few plants, a few events which have significantly probability that we currently see in Surry. That is something which I think will come out within the next twelve months.

Furthermore,if the recommendations of the ACR Sub-committee on Safety Goals are accepted by your Commission, the more stringent NRC proposals do not appear to be justified. Again, Mr. Gibbs addressed that very fully.

Accordingly, we would recommend that the NRC limit its requirements in the short term to the above outlined and focus on what is important, and defer its more stringent prpposals for 12 to 1[illegible] months until many on-going public risk assessments are completed and the safety goal has had a more widespread review. Thank you very much.

CHAIRMAN AHEARNE: When you say focus on efforts on the outlying plants --

DR. WALL: Outlying events on selected plants.

COMMISSIONER HENDRIE: It seems to me that the essence of the difference on the ATWS frequencies, there is about a factor of ten difference in the assumption of the ting rate, and I am not quite sure what the overall average in transient rate would be on the staff side, but I suspect about five, and yours is a little under two, a factor of about two and-a-half.

DR. WALL: Except for Westinghouse.

COMMISSIONER HENDRIE: I am just trying to pick some characteristic of the overall set. Those differences then propagate through and result in a factor of ten difference in the estimates of ATWS occurrence rates.

MR. LELLOUCHE: Ten to twenty. In my analysis of comparative risk I used the staff's numbers for ATWS frequencies, I did not use any other numbers. I compared only the staff number risk. If we had used other numbers, historical numbers, that risk ratio would have been much larger.

COMMISSIONER HENDRIE: I guess the other comment I would like to make - not knowing quite where it fits in the discussion, actually - is that the most prominent recent contribution to ATWS experience for the scrap discharge volume at Brown's Ferry didn't have very much to do with the electrical portion of the system.

CHAIRMAN AHEARNE: That was a comment.

COMMISSIONER HENDRIE: I am not sure what to infer from that, you understand.

MR. LELLOUCHE: Neither was a failure to scram in the ATWS sense in that the residual power level was in the neighborhood of a very few percent. Had it occurred from 100 percent power, it would have still been in the neighborhood of approximately ten percent, which could have been handled by the condenser. It was not an ATWS, it was not even an approximate ATWS.

COMMISSIONER HENDRIE: I recognize that. I regard it not as having particular significance as a piece of data with regard to ATWS frequency, but rather a piece of interesting data with regard to scram system reliability or unreliability.

MR. LELLOUCHE: I agree with you.

COMMISSIONER HENDRIE: I did feel compelled to comment since your testing rate is one which is based on the testing of the electrical portion of the system.

COMMISSIONER BRADFORD: When you say it was not an ATWS or even an approximate ATWS, are you assuming that there could not have been more water held up in the system which would thereby have defeated the scram to a greater extent?

MR. LELLOUCHE: That would have been a different event. I am not suggesting that. I am saying that the event which occurred was not an ATWS, number one. If the event which occurred had occurred at 100 percent power, it would have still not been an ATWS because the residual power level would have been approximately ten percent, and the bypass to the condenser would have handled it. An ATWS by definition, I presume, is one in which we have a pressure spot of some nature of significance, one in which in a BWR the torus starts to overhead significantly. An ATWS in which nothing happens is a benign ATWS. The words are very difficult.

The events could have occurred slighly differently, obviously, in which it would have been worse or better; there is no argument to that, you are correct. The event which did occur, in fact, was not an ATWS in any classical sense of the meaning of the word.

CHA RMAN AHEARNE: Thank you very much, gentlemen. We have certainly, as Mr. Hendrie said, had an informative afternoon session, and I am sure we will have further dialog with our staff.

MR. LELLOUCHE: Thank you, sir.

CHAIRMAN AHEARNE: We stand adjourned.

(Whereupon, at 4:20 p.m. the meeting of the Commission adjourned, to reconvene subject to the call of the Chair.)

NUCLEAR REGULATORY COMMISSION

This is to certify that the attached proceedings before the

COMMISSION MEETING

in the matter of: Presentation by GE, AIF, and G. lellouche (EPRI) on anticipated Transients without SCRAM (ATWS)

Date of Proceeding: October 28, 1980

Docket Number:

Place of Proceeding: October 28, 1980

were held as herein appears, and that this is the original transcript thereof for the file of the Commission.

M. E. Hansen

Official Reporter (Typed)

M. E. Hansen

Official Reporter (Signature)

ATWS:

A REVIEW

BY

G. S. LELLOUCHE

October 28, 1980

VIEW GRAPHS

REACTOR YEARS OF EXPERIENCE

NRR STAFF	659 (1978)
EPRI	900 (1980)

TESTING RATE

NRR STAFF	12/YEAR
EPRI	100/YEAR

APPENDIX 1

THE REACTOR PROTECTION SYSTEM: TESTING AND FUNCTION

NUREG 0460 assumes that 12 tests of the electrical system are performed per year. The EPRI studies indicate that this is in error by at least a factor of 8.

The reactor protection system consists of sensors, logic, bistables, actuators, and breakers. In BWR's the signal proceeds from the sensor through redundant lines to a pair of actuating valves. The PWR systems are more varied at the breaker end consisting of logic systems requiring one out of two (1/2), two out of four (2/4), or a still more complex 8 breaker system (in four pairs of two with a 1/2 followed by a 2/4) to actuate rod motion.

In analyzing actual plant procedures it is necessary to determine the number of trip levels in the plant, their redundancies, and their testing rates. In order to apply this information to predicting scram unavailability it is necessary to determine which trip levels are reached in any transient of significance.

Consider the four plant types individually. The trip level, redundancies, and testing frequencies are as follows:

BWR's

Scram Signals	No. of Channels	Test Frequency
APRM Highflux	4	Weekly
High Main Steamline Radiation	4	Weekly
High Pressure in Vessel	4	30 days
High drywell pressure	4	30 days
MSIV	4	30 days
Turbine Control Valve	4	30 days
Turbine Stop Valve	4	30 days
Others		

AVERAGES ABOUT 5/week

Westinghouse (Senser to .Bistable)

Scram Signals	No. of C.annels	Test Frequency
High Flux	4	Each 28 days
Overtemperature	4	Each 28 days
Overpower ΔT	4	Each 28 days
Low reactor Coolant flow	3/loop	Each 28 days
Low Pressurizer Pressure	4	Each 28 days
High Pressurizer Pressure	4	Each 2C days
High Pressurizer Level	3	Each 28 days
	Average ~	6/week
Bistable to Actuator	6 (2/4)	Each 28 days
Breakers	2 (1/2)	Each 28 days

B & W (Sensor to Bistable)

Scram Signals	No. of Channels	Test Frequency
Power range high flux	4	Each 30 days
Pressure Temperature	4	Each 30 days
Reactor Coolant Temperature	4	Each 30 days
High reactor pressure	4	Each 30 days
Low reactor pressure	4	Each 30 days
Others	Average	6/week

Bistable to Breaker	4 (2/4)	Each 30 days

C.E. (Sensor to Bistable)

Scram Signals	No. of Channels	Test Frequency
High flux	4	Each 30 days
R.C. Flow	4	Each 30 days
Low pressurizer pressure	4	Each 30 days
High pressurizer Pressure	4	Each 30 days
Steam Generator Level	4	Each 30 days
Steam Generator Pressure	4	Each 30 days
Others		

Averages ~ 6/week

Logic 40

Logic trip relays 24 (includes breakers in pairs) each 30 days

Trip Breakers (in pairs,any 1/2 any 2/4) 8 each 30 days

With very few exceptions (and from EPRI NP801 these have very low frequencies). ATWS transients reach at least two diverse trip levels. The following indication of trip levels come from vendor documents.

BWR TRIP LEVELS

Transient	Trip Levels Reached
Loss of Condensor Vacuum	Stop Valves, Flux, Vessel Pressure
MSIV closure (all loops)	Flux, Vessel Pressure, Stop Valves
Turbine Trip	Same
Generator Trip	Same
Pressure Regulator Failure	Flux, Vessel Pressure
Loss of Feedwater Flow	Low Water Level, Isolation Va es Flux, Vessel Pressure

TRIP LEVELS REACHED DURING W ATWS TRANSIENTS

Transient	RPS Trip Due To
Loss of Load	Turbine trip High Pressurizer Pressure Over temperature ΔT
Loss of Feedwater	Turbine Trip Over temperature ΔT High Pressurizer Pressure
Loss of Offsite Power	Undervoltage Underfrquency Over temperature ΔT Over power ΔT Others
Rod Withdrawal	High Flux Over temperature ΔT Over power ΔT Pressurizer high level

SUMMARY OF TESTING RATES FOR EACH REACTOR

BWR's

Depending on Transient	100-200/year

PWR's

Sensors to Bistable

Depending on transient	100-200/year

Bistable to Actuator

W	78 /year
B&W	48/year
C.E.	480/year

Breakers

W	24/year
B & W	48/year
C. E. (Direct test)	96/year
C. E. (with Logic Trip Relays)	288/years

SCRAM UNAVAILABILITY / DEMAND

STATISTICAL CONFIDENCE LEVEL	5%	50%	95%
HISTORICAL DATA			
WITHOUT KAHL	2.8×10^{-7}	3.8×10^{-6}	1.7×10^{-5}
WITH KAHL	2×10^{-6}	9×10^{-6}	2.4×10^{-5}
FAULT TREE SYNTHESIS			
BWR	5.1×10^{-7}	2.3×10^{-6}	2×10^{-5}
PWR	1.7×10^{-6}	4.2×10^{-6}	1.1×10^{-5}
UPPER BOUND EFFECT OF RODS = 10^{-6}			
BWR	$< 10^{-6}$	$< 3 \times 10^{-6}$	$\sim 2.1 \times 10^{-5}$
PWR	$< 3 \times 10^{-6}$	$< 5 \times 10^{-6}$	$\sim 1.2 \times 10^{-5}$

NRC CALCULATION OF ROD & EFFECT

EASTERLING, VESELY	BWR	10^{-7}	AT 99% S-C
EASTERLING	PWR	2×10^{-6}	AT 99% S-C

Limiting Transients for ATWS*

I. Babcock & Wilcox

A. Loss of offsite power (LOOP)

B. Total loss of feedwater (LOF)

C. Transients leading to LOF (LOL)

II. Combustion Engineering

A. 2560 MWt Core

1. Uncontrolled rod withdrawal (CEA)
2. Partial loss of feedwater (PLOF)
3. Loss of load (LOL)
4. Total loss of feedwater (LOF)

B. 3800 MWt Core

1. Uncontrolled rod withdrawal
2. Partial loss of primary coolant flow (PPCF)
3. Loss of load
4. Total loss of feedwater

III. Westinghosue (No transient yields results of significance but the most limiting transients are the following)

A. Loss of load

B. Total loss of feedwater

IV. General Electric

Any transient leading to excessive pool temperatures (GE)

---

* These transients have been specified by NRC in WASH 1270 and the Status Reports as being those which lead to excessive pressures.

Correspondence Between Significant ATWS
Transients and Plant Transient Data

ATWS Transient		Plant Transient
PWR		
PPCF	# 1*	Loss of RCS (1 Loop)
CEA	# 2	Uncontrolled Rod Withdrawal
PLOF	#15	Loss or Reduction in Feedwater Flow (1 Loop)
LOF	#16	Total Loss of Feedwater Flow (All Loops)
LOL	#18	Closure of All MSIV
	#24	Loss of Condensate Pumps (All Loops)
	#25	Loss of Condensor Vacuum (LCV(
	#33	Turbine Trip (TT)
	#34	Generator Trip (GT)
LOOP	#35	Loss of Station Power
BWR	# 1	Load Rejection
	# 3	Turbine Trip
	# 5	MSIV (All Loops)
	# 8	Loss of Condenser Vacuum
	# 9	Pressure Regulator Fails Open
	#10	Pressure Regulator Fails Closed
	#20	Feedwater, Increasing Flow at Power
	#24	Feedwater, Low Flow
	#31	Loss of Offsite Power
	#32	Loss of Auxiliary Power

* This number refers to the detailed transient frequencies presented in EPRI NP 801

Reactor Median Transient Initiation

Frequencies Relevant for ATWS

		Events/Year
I.	Babcock & Wilcox	
	1) LOOP	0.27
	2) LOF	0.07
	3) LOL	1.11
		Sum =1.45
II.	Combustion Engineering	
	a) 2560 MWt Core	
	1) CEA	0.02
	2) PLOF	0.45
	3) LOL	1.11
	4) LOF	0.07
		Sum =1.65
	b) 3800 MWt Core	
	1) CEA	0.02
	2) PPCF	0.13
	3) LOL	1.11
	4) LOF	0.07
		Sum =1.23
III.	Westinghouse (none of significance, but those most limiting are)	
	1) LOL	1.11
	2) LOF	0.07
		Sum =1.18
IV.	General Electric	Sum =3.52

NEW STAFF TRANSIENTS

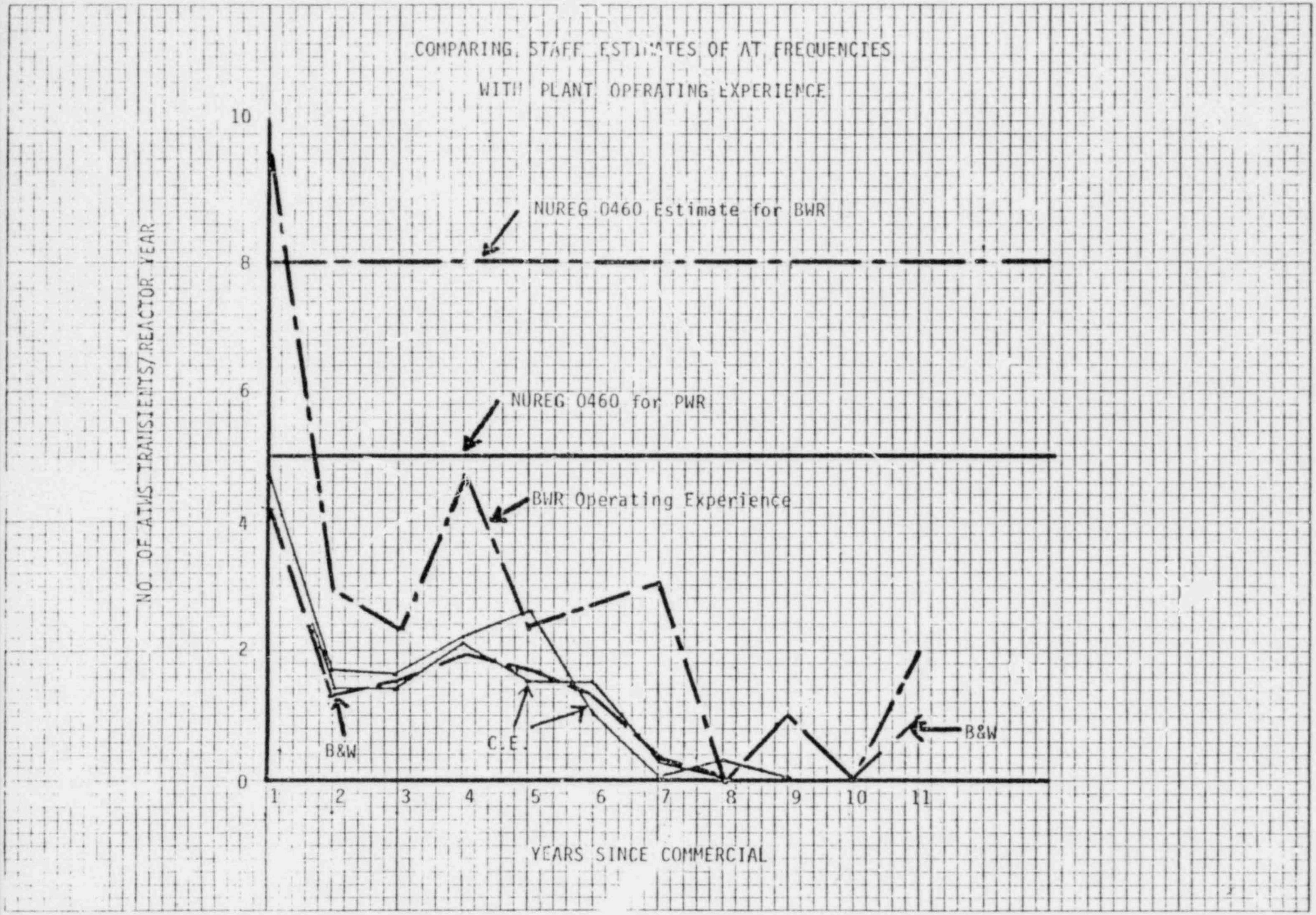
PWR

	FREQUENCIES		
	TOTAL	25% POWER	SIGNIFICANCE
STUCK VALVES	0.03	?	NONE
SAFETY INJECTION	0.04	0.01	NONE
FEEDWATER FLOW INSTABILITY	1.16	1.02	MAYBE
LOSS OF CIRCULATING WATER	0.07	0.08	YES
LOSS OF POWER TO NECESS. SYS.	0.21	0.05	NONE
LOW SECONDARY PRESSURE	0.06	0.04	NONE

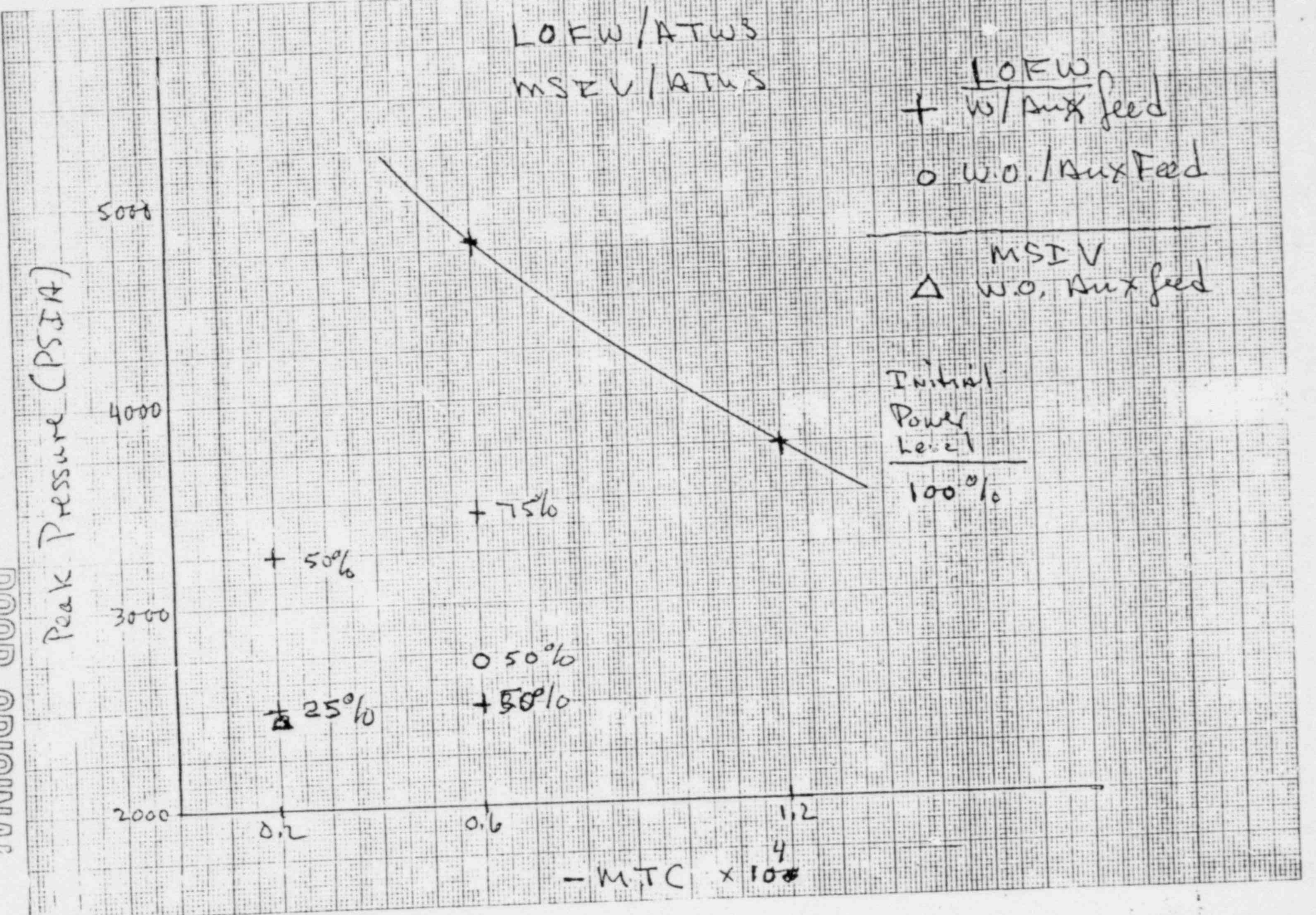
BWR

STUCK VALVES(NO AUTO.SCRAM)	0.2	0.13
BYPASS VALVE FAILS OPEN	0.04	0.0
MSIV (1 LOOP)	0.08	0.07
LOSS OF FEEDWATER HEATING	0.02	0.0

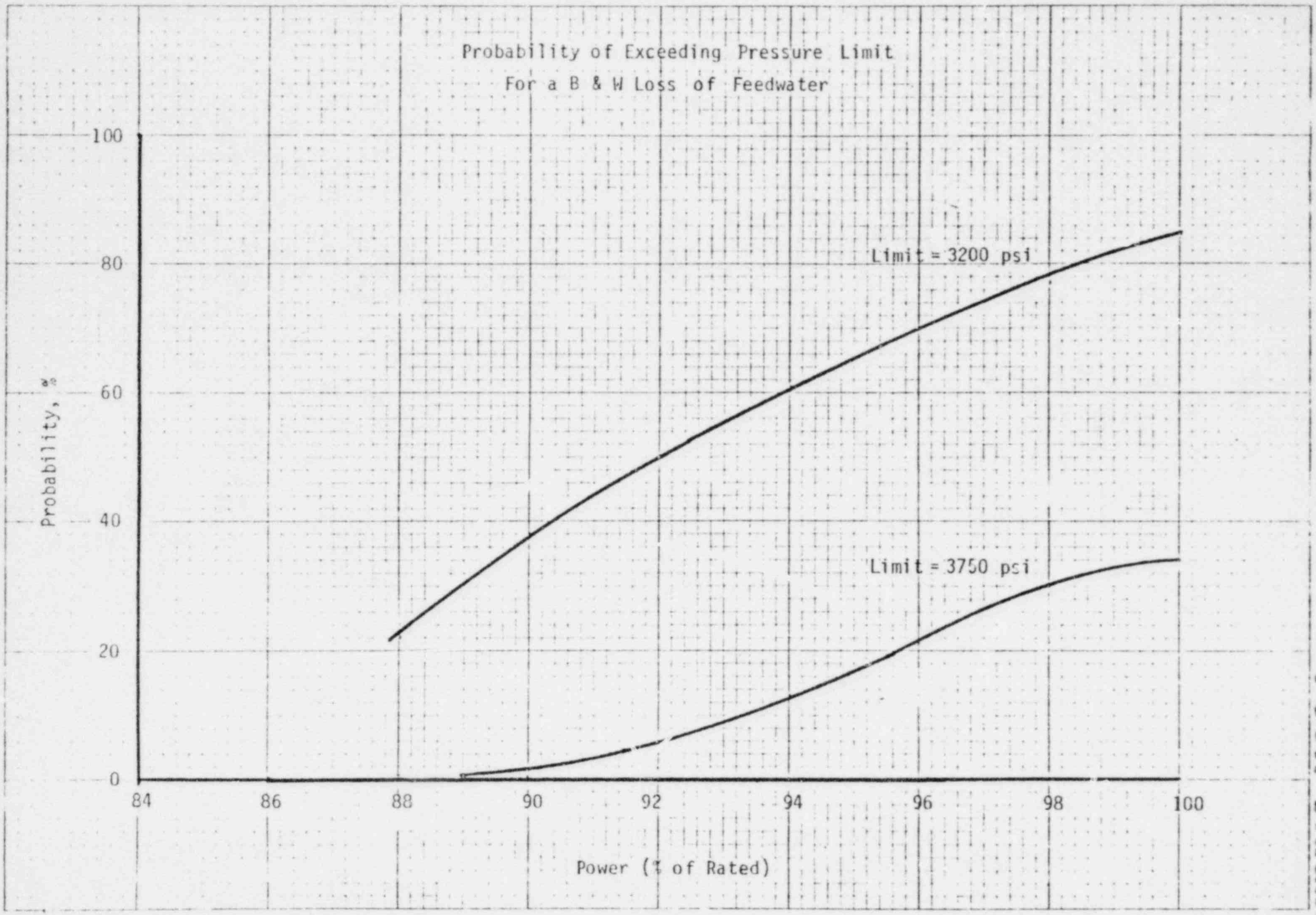
GSL:NS 10/24/80


COMPARING STAFF ESTIMATES OF AT FREQUENCIES
WITH PLANT OPERATING EXPERIENCE
NUREG 0460 Estimate for BWR
NUREG 0460 for PWR
BWR Operating Experience
B&W
C.E.
B&W
NO. OF ATWS TRANSIENTS/REACTOR YEAR
YEARS SINCE COMMERCIAL
10
8
6
4
2
0
1
2
3
4
5
6
7
8
9
10
11

C.E. 2611 MW PLANT



POOR ORIGINAL



POOR ORIGINAL

TABLE IX

Effect of Plant Generation on Transient Event Rates

Plant Type		Year of Operation 1	2	3
PWR's	Greater than 6 years old	19.7	19.7	12.6
	Less than 6 years old	16.9	10.3	7.8
BWR's	Greater than 6 years old	20.3	5.5	5
	Less than 6 years old	23.4	7	5

TABLE X

LWR Applicable At Frequencies

PWR	0.64 (0.24 w/100% bypass)
BWR	3.52 (1.22 w/> 30% bypass)
LWR	1.68 (0.60 w/appropriate bypass)

TABLE XI

Annual Frequency of ATWS (Pr(ATWS))

PWR	3.2×10^{-6} (1.2×10^{-6} w/bypass)
BWR	1.8×10^{-5} (6.1×10^{-6} w/bypass)
LWR	8.4×10^{-6} (3×10^{-5} w/bypass)

MEDIAN ATWS FREQUENCIES

NRR STAFF	W	CE	B&W	GE
0460 VOL 1-3	2×10^{-4}	2×10^{-4}	2×10^{-4}	2×10^{-4}
0460 VOL 4	$10^{-6}-8 \times 10^{-5}$	8×10^{-5}	8×10^{-5}	2×10^{-4}
EPRI	$\ll 10^{-6}$	$6-9 \times 10^{-6}$	8×10^{-6}	1.8×10^{-5}

Figure 6

ATWS Event Sequences During
the Fuel Cycle

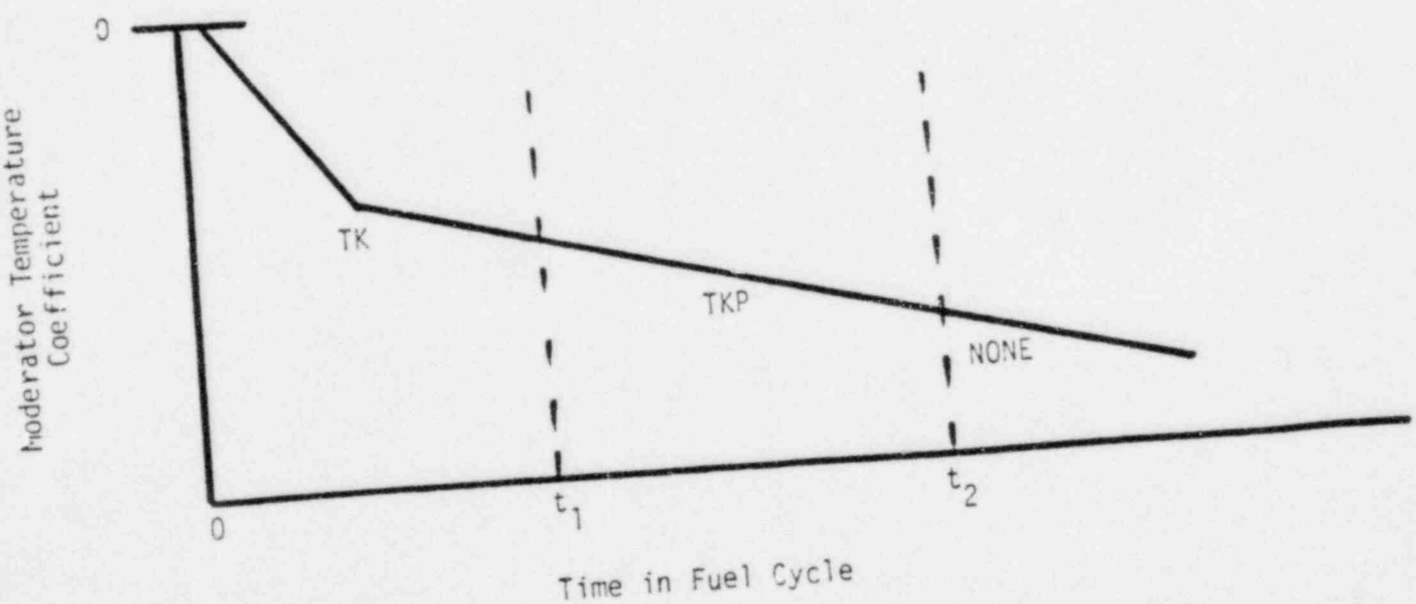
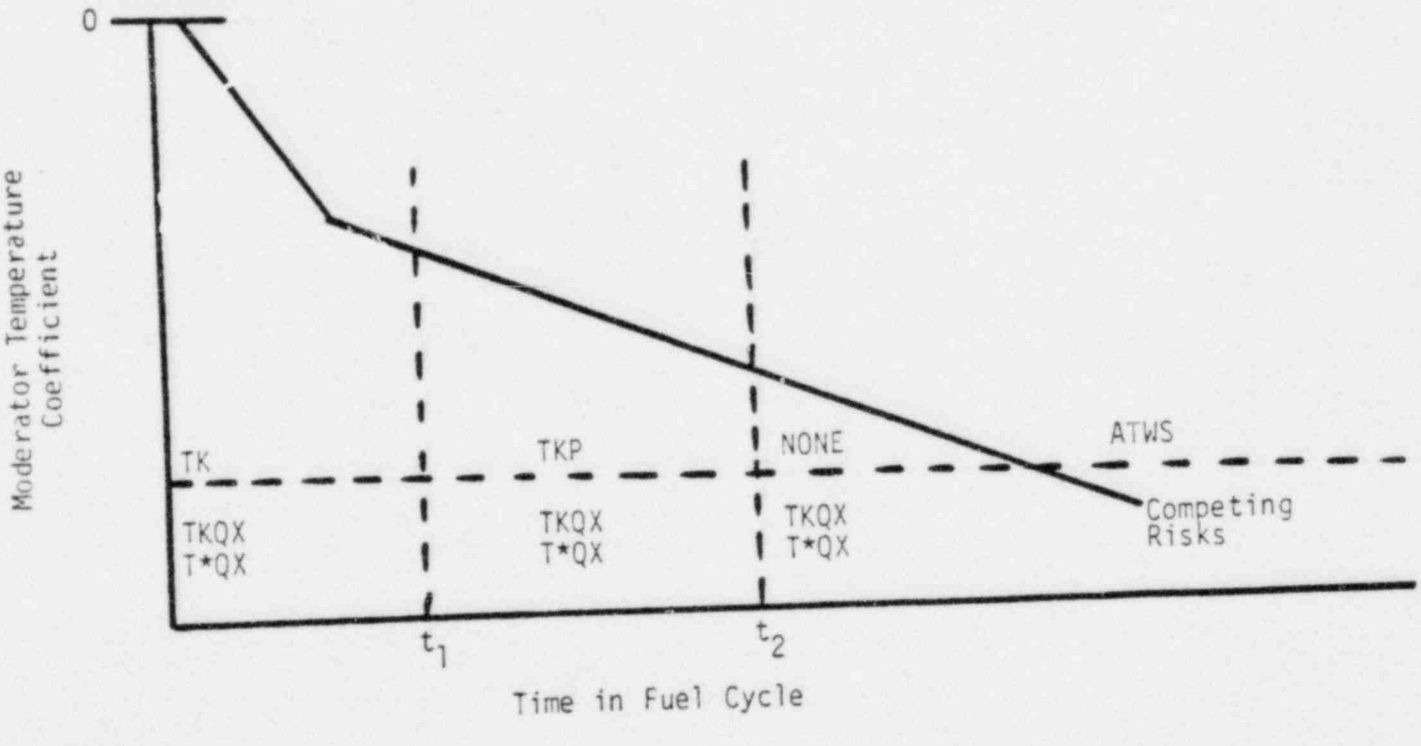


Figure 7



0
Moderator Temperature Coefficient
TK
TKP
NONE
ATWS
TKQX
T*QX
TKQX
T*QX
TKQX
T*QX
Competing Risks
t1
t2
Time in Fuel Cycle

APPENDIX

List of (Stuck Open) Pressurizer Valves

1. Palisades, 1971, PORV
2. Oconee 1, 1973, Block Valve
3. Oconee 3, 1975, PORV
4. Davis Besse, 1977, PORV
5. TMI-2, 1978, PORV
6. Cook 3, 1978, PORV
7. Ft. Calhoun, 1979, 2 PORV
8. TMI-2, 1979, PORV
9. Crystal River, 1980, ~~Safety~~ POR Valve

C - Failure of Containment Spray Injection System

D - Failure of Emergency Core Cooling Injection

F - Failure of Containment Spray Circulation System

G - Failure of Containment Heat Removal System

H - Failure of Emergency Core Cooling Recirculation System

$0 - t_1$: TK >> TKQX

$t_1 - t_2$: TKP < to << TKQX

t_2 onward: TKQX is the total risk

Westinghouse	B&W	Combustion
• Loss of external Load	• Loss of External Load	• Loss of External Load
• Turbine Trip	• Turbine Trip	• Turbine Trip
• Loss of Normal Feedwater	• Loss of Normal Feedwater	• Loss of A.C. Power
• Loss of all A.C. Power	• Cold Pressurization	• Cold Pressurization
• Cold Pressurization		

$S_2D = 9\times10^{-6}$/Reactor Year

$S_2F = 1\times10^{-7}$/Reactor Year

$S_2G = 9\times10^{-8}$/Reactor Year

$S_2H = 6\times10^{-6}$/Reactor Year

$S_2C = 2\times10^{-6}$/Reactor Year

$D = 9\times10^{-3}$/demand

$F = 1\times10^{-4}$/demand

$G = \cancel{1\times10^{-8}}$/demand 9×10^{-5}

$H = 6\times10^{-3}$/demand

$C = 2\times10^{-3}$/demand

hence that

$$(T^*Q)\ (D + F + G + H + C) \equiv T^*QX = 5\times10^{-4}$$

TK vs. $TKQX + T^*QX$

Since $K \ll 1$, we can neglect (for now) $TKQX$. NUREG 0460[+] estimates

$$TK = 1.6 \times 10^{-4}\text{/Reactor Year}$$

Since $T^*QX = 5 \times 10^{-4}$/Reactor Year, valve failures to reset already dominate ATWS risk hence any additional valves can only increase the risk.

C - Surry does not contain fan coolers; for plants with fans as well as sprays C is negligable.

D - Plant changes should not significantly affect this parameter.

F - Plant variations imply that F and H are coupled hence that F should not be called out separately; since H >> F this does not impact.

G,H - Plant variations do not indicate that these should change signifi- significantly.

With these considerations we recalculate T^*QX for non-Surry type of plants to be

$$X = 1.6 \times 10^{-2}\text{/demand}$$

and

$$T^*QX = 5 \times 10^{-4}\text{/reactor year}$$

hence for non-Surry plants ATWS is still only ¼ of T^*QX hence additional valves will increase risk. If we use $t_1 \approx 0.4$ the Q failure core melt probability dominates ATWS by a factor of 10.

Release as Equivelent Iodine-131

$\frac{PWR-3}{PWR-5}$ = 20

$\frac{PWR-3}{PWR-7}$ = 20,000

$TK = 1.6x10^{-4}$/year

The containment failure modes for ATWS by Category are then

Category 7 $TK\text{-}\varepsilon = 1.6x10^{-4}$

Category 5 $TK\text{-}\beta = 6.4x10^{-7}$

The total ATWS risk is

$(TK\text{-}\varepsilon)\ C_7 + (TK\text{-}\beta)\ C_5$

$(T^*Q(D+H)\text{-}\alpha + T^*Q(F+C+G)\text{-}\delta + TKQ\text{-}\alpha)\ C_3$

and the risk ratio is

$$\frac{\text{Competing Risk}}{\text{ATWS Risk}} = \frac{\{T^*Q[(D+H)\text{-}\alpha+(F+C+G)\text{-}\delta]+TKQ\text{-}\alpha\}C_3}{[TK(\varepsilon C_7+\beta C_5)]\,t_1}$$

Quantifying this relation

$$\frac{\text{Completing Risk}}{\text{ATWS Risk}} = \frac{6.6 \times 10^{-5}\ C_3}{6.4 \times 10^{-5}\ (.004\ C_5 + C_7)}$$

$$\simeq 5000$$

Neglecting C and F for non surry types of plants

$$= \frac{7.2 \times 10^{-6}\ C_3}{6.4 \times 10^{-5}\ [.004\ C_5 + C_7]}$$

$$= 522$$

CRYSTAL RIVER

1. T^*Q is a B_y (small-small) LOCA

2.

B_y Seg #	MELT Prob given B_y	Type
2	1.3×10^{-2}	ECC Recirc. Fails (H)
6	1.2×10^{-3}	Bldg Spray and ECC Recirc (F+H
9	7.2×10^{-3}	ECC Injection Fails (D)
20	2.6×10^{-3}	Bldg Spray & Recirc Fails (C+H
23	1.0×10^{-2}	Bldg Spray and ECC fails (C+D

$$\Sigma = 3.4 \times 10^{-2}$$

$$T^*QX = 1.02 \times 10^{-3}$$

Risk

$$\frac{\text{Non ATWS}}{\text{ATWS}} = \frac{(S_2+S_{23})(\alpha C_1+\gamma C_2)+(S_2+S_6+S_{20}+S_{23})\varepsilon C_6 + S_9\,\varepsilon C_7}{T k t_1(\beta C_5 + \delta C_7)}$$

$$= 10000$$

POOR ORIGINAL

PRESENTATION

TO

U.S. NUCLEAR REGULATORY COMMISSION

ON

ANTICIPATED TRANSIENTS WITHOUT SCRAM

OCTOBER 28, 1980

DR. D. CLARK GIBBS

My name is Clark Gibbs. I am Director of Nuclear Activities for Middle South Services and Vice President of Middle South Energy, Inc., the owner of the Grand Gulf Nuclear Station. I am here today as Chairman of the AIF Committee on Reactor Licensing and Safety. I am also a member of the AIF Policy Committee on Nuclear Regulation and the EEI Executive Advisory Committee on Nuclear Power. The statement on ATWS that I shall make before you today has the endorsement of these AIF and EEI committees as well as the members of the APPA Nuclear Power Task Force which currently own and operate nuclear power plants on their systems.

I will be reading my prepared presentation to you because of the organizations which I represent here and the need for their considered review of my remarks. Should you have questions during these prepared remarks, do not hesitate to interrupt me. I am joined here today by Fred Stetson of the AIF staff, Jerry Sorensen, Chairman of the AIF ATWS Subcommittee, and Dr. Anthony Buhl, Vice President of Technology for Energy Corporation, who will assist as necessary in dealing with your questions. Also present are others from the industry whom I may call upon should the need arise.

Both the NRC and the industry are vitally interested in the safety of nuclear power, largely for the same reasons. Those of us who advocate continued and expanded use of nuclear

POWER HAVE GROWN ACCUSTOMED TO THE ATTENTION TO DETAIL, ENERGY, AND COMMITMENT THAT THE ASSURANCE OF NUCLEAR SAFETY REQUIRES. WE WELL UNDERSTAND THE POTENTIAL CONSEQUENCES OF ERRORS IN JUDGEMENT ON PUBLIC ACCEPTANCE, UNIT AVAILABILITY, AND COST COMPARISONS WITH ALTERNATIVES. THOSE OF US WHO ARE OWNERS OF THESE PLANTS ARE KEENLY AWARE OF THE IMPORTANCE THAT OUR RATEPAYERS WHO LIVE IN THE ENVIRONS OF OUR PLANTS ATTACH TO NUCLEAR SAFETY. WE HAVE NOT FAILED TO OBSERVE AS WELL THE HIDEOUS FINANCIAL IMPACT ATTENDANT WITH AN EVENT WHICH COMPROMISES OUR ABILITY TO PROVIDE ADEQUATE COOLING FOR THE REACTOR CORE. WE HAVE EVERY REASON TO BE THE MOST COMMITTED TO NUCLEAR SAFETY OF ANY ORGANIZATION PARTICIPATING IN ITS USE.

IT IS FROM THAT PERSPECTIVE WHICH WE VIEW THE ATWS ISSUE, ONE WHICH HAS CONFOUNDED OVER 10 YEARS OF ATTEMPTED RESOLUTIONS. WE BELIEVE THAT THE UNDERLYING REASON FOR THE INORDINATE LENGTH OF TIME AND EFFORT THAT HAS ALREADY BEEN EXPENDED ON THIS SUBJECT AND WHICH HAS BEEN FREQUENTLY SPICED WITH ACERBIC DIALOG IS THAT IT IS AN UNPRECEDENTED ATTEMPT TO PROVIDE PROTECTION FOR A SINGLE EXTREMELY SMALL PROBABILITY EVENT, FROM AMONG A HOST OF OTHERS WHICH MAY HAVE A GREATER PROBABILITY OF OCCURRENCE AND FOR WHICH THE CONSEQUENCES ARE LIKELY TO BE MORE SEVERE. WE WISH TO ENHANCE AS NECESSARY THE SAFETY AND OPERABILITY OF OUR PLANTS IN A FASHION WHICH IS SELF CONSISTENT, AND OBJECTIVELY ALLOCATES OUR RESOURCES TOWARD THE

ACHIEVEMENT OF A WELL UNDERSTOOD SAFETY GOAL BASED UPON A FIRM FOUNDATION OF ANALYSIS OF BENEFITS AND COMPETING SOCIETAL RISKS. IN FACT, IT APPEARS TO US THAT THE TREATMENT OF THIS SUBJECT BY THE NRC STAFF HAS BEEN CLEARLY OVERTAKEN BY THE EVENTS WHICH HAVE OCCURRED SINCE THE ACCIDENT AT TMI. THE SPECIFIC EVENTS TO WHICH I ALLUDE ARE THE RENEWED INTEREST IN THE ESTABLISHMENT OF QUANTITATIVE SAFETY GOALS, THE ONGOING AND PLANNED PROBABILISTIC ASSESSMENT STUDIES AND THE PLANNED DEGRADED CORE RULEMAKING. IT IS FROM THESE ACTIVITIES THAT WE PROPOSE THAT THE ULTIMATE RESOLUTION OF ATWS BE DERIVED.

IN THE INTEREST OF EXPANDING UPON THIS PROPOSAL WE SUGGEST THAT THE FIRST PREREQUISITE FOR A FINAL ATWS RESOLUTION IS THE DEFINITION OF A SAFETY GOAL FOR NUCLEAR POWER PLANT REGULATION. THE OPTIMUM ATWS RESOLUTION INVOLVES THE REDUCTION OF RISKS THAT ARE ALREADY VERY SMALL. SINCE IT IS IMPOSSIBLE TO REDUCE RISKS TO ZERO, WE CONTINUE TO BE CONFRONTED WITH THE QUESTION, "HOW SAFE IS SAFE ENOUGH?" ALTHOUGH, OF NECESSITY, THE LACK OF A SAFETY GOAL HAS NOT PRECLUDED RULEMAKING IN THE PAST, IT WOULD BE UNWISE TO IGNORE SAFETY GOAL GUIDANCE THAT SHOULD SOON BE AVAILABLE. RECENT RECOGNITION THAT SUCH GUIDANCE IS ESSENTIAL SUGGESTS THAT IT WILL BE AVAILABLE IN TIME TO GUIDE A FINAL ATWS RESOLUTION. I SHOULD POINT OUT AT THIS JUNCTURE THAT THE AIF COMMITTEE ON REACTOR LICENSING AND

SAFETY HAS RECENTLY COME FORWARD WITH A PROPOSED SAFETY GOAL BEFORE THE ACRS WHICH HAS RECEIVED SUPPORT WITHIN THE INDUSTRY.

A SECOND PREREQUISITE FOR A FINAL ATWS RESOLUTION IS FURTHER WORK ON PROBABILISTIC RISK ASSESSMENT ANALYSIS. THE LAST COMPREHENSIVE PRA WHICH HAS BEEN PERFORMED AND WIDELY CIRCULATED AND WHICH TREATS ATWS AMONG ALL THE OTHER EVENTS THAT CAN LEAD TO DEGRADED CORE COOLING CONDITIONS WAS WASH-1400. THAT STUDY SUGGESTED THAT THE RISK FROM ATWS EVENTS IN LWR'S WAS SMALL. OTHER NRC STUDIES SUCH AS THE FOUR VOLUMES OF NUREG 0460 HAVE TREATED ATWS IN GREATER DETAIL THAN WASH-1400 BUT HAVE DONE SO IN ISOLATION OR HAVE COMPARED A REVISED ATWS RISK WITH UNMODIFIED WASH-1400 VALUES FOR COMPETING RISKS. THIS IS CLEARLY INAPPROPRIATE AND PARTICULARLY SO IN VIEW OF THE SIGNIFICANT WORK UNDERWAY AND PLANNED TO EXPAND THE BASE OF OUR KNOWLEDGE IN THE AREA OF PRA. WITHIN THE INDUSTRY A GROWING NUMBER OF PRA EVALUATIONS ARE SCHEDULED FOR COMPLETION IN THE NEAR FUTURE THAT WILL PROVIDE INSIGHTS ON ATWS.

THE THIRD PREREQUISITE FOR FINAL ATWS RESOLUTION IS THE INTEGRATION OF ATWS INTO THE PLANNED DEGRADED CORE RULEMAKING. THIS RULEMAKING WILL DETERMINE WHETHER AND TO WHAT EXTENT DEGRADED CORE OR CORE MELT ACCIDENTS MUST BE CONSIDERED IN SAFETY ANALYSES. THE END RESULT OF THIS PROCESS MAY BE A RULE THAT WILL AMEND 10 CFR 50 TO REQUIRE CHANGES IN PLANT DESIGN OR

PROCEDURES THAT WILL IMPROVE THE CAPABILITY OF LIGHT WATER REACTORS TO PREVENT, RESPOND TO, OR ACCOMMODATE THE EFFECTS OF ACCIDENTS RESULTING IN A DEGRADED REACTOR CORE.

AS NOTED ABOVE, THE INDUSTRY DOES NOT BELIEVE THAT FINAL ATWS RESOLUTION CAN BE ACHIEVED INDEPENEDENT OF THE DEGRADED CORE RULEMAKING. A SYSTEMATIC SAFETY EVALUATION OF A NUCLEAR POWER PLANT SHOULD CONSIDER ALL THE SEQUENCES AND SUGGESTED MODIFICATIONS IN PERSPECTIVE. IN THIS MANNER WE CAN DIRECT OUR ATTENTION AND RESOURCES TO THE DOMINANT SEQUENCES THAT IMPACT SAFETY AS WELL AS TO EVENTS THAT COULD RESULT IN OTHER SEVERE CONSEQUENCES. BECAUSE THE SAME ISSUES AND FACTS ARE CRUCIAL TO EACH, ATWS IS SIMPLY A SUB PART OF THE DEGRADED CORE MATTER; WE RECOGNIZE THAT THE RISK OF ATWS, TO THE EXTENT THAT THERE IS ANY SIGNIFICANT RISK, IS ONE OF DEGRADED CORE. WE RECOGNIZE THAT ATWS IS ONE RELATIVELY LOW-PROBABILITY EVENT AMONG MANY THAT COULD CONCEIVABLY LEAD TO A DEGRADED CORE. ACCORDINGLY, THERE SEEMS TO BE NO SOUND REASON FOR SEEKING FINAL ATWS SOLUTIONS FOR PLANTS IN ISOLATION FROM OTHER DEGRADED CORE EVENTS.

WE WOULD PREFER TO AVOID CONTINUED DIALOG ON ATWS INDEPENDENTLY, AND THEREFORE PROPOSE THAT THE MATTER BE DISPOSED OF NOW IN A FASHION WHICH IS SUPPORTED BY THE RECORD AND RESULTS IN SUBSTANTIAL REDUCTION OF ATWS RISK. THE STAGE HAS BEEN SET TO TREAT THE RESIDUAL ATWS RISK IN THE DEGRADED

CORE COOLING RULEMAKING IN A FASHION WHICH WILL BE ACCEPTABLE TO THE INDUSTRY AND IN PARTICULAR, TO THE OWNERS OF THESE PLANTS.

THERE REMAINS THE QUESTION OF WHAT CAN AND SHOULD BE DONE NOW. THE STAFF HAS RECENTLY PROPOSED AN ATWS RULE AND REGULATORY GUIDE CONTAINED IN SECY-80-409. YOU HAVE ALSO BEEN SERVED WITH A PETITION FOR RULEMAKING BY THE ATWS UTILITY GROUP REPRESENTING 20 DOMESTIC ELECTRIC UTILITY COMPANIES. THE TWO PROPOSED RULES ARE QUITE SIMILAR INSOFAR AS SPECIFIC SHORT TERM HARDWARE REQUIREMENTS ARE CONCERNED. BEYOND THAT, THEY DIVERGE. IN THE LONGER TERM, THE STAFF PROPOSES TO SPECIFY CRITERIA RATHER THAN MITIGATING HARDWARE. WE BELIEVE THIS IS A SIGNIFICANT POSITIVE STEP AND THAT A FINAL RULE WHICH MAY EVOLVE AS A PRODUCT OF THE DEGRADED CORE RULEMAKING SHOULD RIGHTFULLY ADDRESS ITSELF TO CRITERIA RATHER THAN HARDWARE.

HOWEVER, THE PROPOSED CRITERIA ARE PREMATURE AND AS A RESULT, DEFICIENT. IN OUR JUDGEMENT, THE STAFF PROPOSALS DO NOT PROVIDE CLOSURE OF THE ATWS ISSUE. THE PROPOSED REGULATORY GUIDE WILL AFFORD TO THE STAFF UNRESTRICTED OPPORTUNITIES FOR IMPOSING FURTHER REGULATORY REQUIREMENTS WHICH WILL INEVITABLY RESULT IN ATWS BECOMING A DESIGN BASIS EVENT FOR STRUCTURES, SYSTEMS AND COMPONENTS WITH IMPLICATIONS FAR BEYOND THAT OF WHICH ANY OF US TODAY ARE CAPABLE OF IMAGINING. THE APPEARANCE OF A NEW DESIGN BASIS EVENT VIRTUALLY GUARANTEES SUBSTANTIAL

IMPACTS ON THE RESOURCES OF BOTH THE NRC AND INDUSTRY FOR MANY YEARS IN THE FUTURE.

THE PROPOSED INTEGRAL PLANT AND SEPARATE EFFECTS TESTING IDENTIFIED IN THE REGULATORY GUIDE ARE BRIEFLY OUTLINED AS TO PURPOSE ONLY. THERE IS NO WAY OF INTELLIGENTLY EVALUATING WHAT IS EXPECTED OF US FROM THESE PURPOSE STATEMENTS AND CERTAINLY NOT IN THE TIME ALLOWED FOR IN THE SCHEDULE WHICH I WILL ADDRESS LATER. FURTHER, THE APPEARANCE OF THESE TESTS IS ADDITIONAL EVIDENCE THAT THE STAFF IS MOVING IN THE DIRECTION OF TREATING ATWS AS A DESIGN BASIS EVENT AFTER THE FASHION OF THE DESIGN BASIS LOSS OF COOLANT ACCIDENT, A PRACTICE WHICH LED TO SOME OF THE UNFAVORABLE FINDINGS OF THOSE CHARGED WITH THE TASK OF EVALUATING NRC'S PERFORMANCE FOLLOWING THE TMI ACCIDENT. THE STAFF PROPOSALS ARE PARTICULARLY DEFICIENT IN THE ASSOCIATED VALUE-IMPACT ANALYSES, PROPOSED SCHEDULE FOR IMPLEMENTATION, AND ATTENTION TO DETAIL WHERE CONTRADICTIONS CLEARLY EXIST ON THE RECORD.

BEGINNING WITH VALUE-IMPACT, THE NUCLEAR REGULATORY COMMISSION HAS ADOPTED A POLICY, "THAT VALUE-IMPACT ANALYSIS WILL BE CONDUCTED FOR ANY PROPOSED REGULATORY ACTIONS THAT MIGHT IMPOSE A SIGNIFICANT BURDEN ON THE PUBLIC (WHERE THE TERM PUBLIC IS DEFINED IN ITS BROADEST SENSE)." CONSISTENT WITH THIS POLICY, THE NRC STAFF HAS ATTEMPTED TO DEVELOP THE REQUIRED VALUE-IMPACT ANALYSIS FOR ATWS. THE STAFF'S EFFORT TO

DATE, HOWEVER, HAS NOT BEEN ADEQUATE. THE MAJOR DEFECTS INCLUDE FIRST, FAILURE TO REALISTICALLY CONSIDER THE CONSUMER IMPACTS ASSOCIATED WITH MAJOR BACKFITS AND EXTENDED OUTAGES THAT WILL INCREASE THE COST OF ELECTRIC POWER. THE STAFF REPORTS IN SECY-80-409 THAT IT IS THEIR JUDGEMENT THAT EXTENDED DOWNTIME REQUIRED TO RETROFIT WILL LIKELY BE MINIMAL. IN VIEW OF THE APPARENT NEED TO PROVIDE ADDITIONAL RELIEF VALVE CAPACITY TO MEET THE ACCEPTANCE CRITERIA OF THE PROPOSED RULE IN B&W AND COMBUSTION DESIGNED PLANTS, THIS STATEMENT IS PROFOUNDLY IN ERROR.

A RECENT STUDY PERFORMED AT DUKE POWER COMPANY INDICATES THAT A MINIMUM OF 31 DAYS OF ADDITIONAL DOWN TIME WOULD BE REQUIRED TO MAKE THE PRESSURIZER MODIFICATIONS ON OCONEE NECESSARY TO PROVIDE THE ADDITIONAL RELIEF PROTECTION MANDATED BY THE ACCEPTANCE CRITERIA ASSUMING <u>ABSOLUTELY NO PROBLEMS</u>, A MOST UNLIKELY ASSUMPTION. THE STUDY FURTHER ESTIMATES THAT THIS UNAVAILABILITY WOULD BE LIKELY TO GROW TO 65 DAYS IF <u>EXPECTED</u> PROBLEMS MANIFEST THEMSELVES SUCH AS DIFFICULTY IN REMOVING THE PRESSURIZER MANWAY, OR REPAIR OF INDICATIONS ON THE NOZZLE WELDS. APPROXIMATELY 360 MAN-REM OF OCCUPATIONAL EXPOSURE WOULD BE INVOLVED ON EACH UNIT. USING 2000 $ PER OCCUPATIONAL MAN-REM AND $200 K PER UNIT PER DAY COST OF REPLACEMENT POWER, WHICH FOR DUKE IS NEARLY ALL COAL, THEY ESTIMATE A $25M IMPACT ON THEIR THREE OCONEE UNITS EXCLUSIVE OF

ENGINEERING AND EQUIPMENT COSTS. IT IS IMPORTANT TO POINT OUT THAT OCONEE OPERATION HAS BEEN RELATIVELY FREE OF FUEL FAILURES AND THEIR RESULTANT EXPOSURES WILL BE CONSIDERABLY BELOW THE AVERAGE WHEN PLANTS WHICH HAVE EXPERIENCED OPERATION WITH FAILED FUEL ARE TAKEN INTO ACCOUNT.

IN ADDITION, THE OCONEE CONTAINMENT IS RELATIVELY UNCONGESTED, MINIMIZING THE INTERFERENCE PROBLEM WHICH WILL BE EXPERIENCED BY OTHERS. ANY OTHER UTILITIES WILL BE REQUIRED TO USE OIL AS A REPLACEMENT FUEL. IT IS THEREFORE JUDGED THAT THE DUKE ESTIMATES PROBABLY REPRESENT A LOWER BOUND ON THE COST OF THIS SINGLE MODIFICATION TO THE CE AND B&W DESIGNED REACTORS. BECAUSE OF THE SENSITIVITY OF THIS ANALYSIS TO COST OF REPLACEMENT POWER AND OUTAGE TIME, THE ULTIMATE RELATIVE IMPACT TO SOME UTILITIES MAY BE A FACTOR OF 5 OR MORE GREATER THAN THAT SUGGESTED BY THE DUKE STUDY.

ANOTHER CONSIDERATION IS THAT THE FULL IMPLEMENTATION OF THE NRC PROPOSED RESOLUTION MAY ALSO REDUCE SYSTEM AVAILABILITY AND RELIABILITY BY MAKING NUCLEAR PLANTS MORE COMPLEX AND, THEREFORE MORE SUBJECT TO MALFUNCTION WHEN EVENTS SUCH AS INADVERTENT INITIATION OF THE AUTOMATIC SLCS ARE TAKEN INTO ACCOUNT.

A SECOND DEFICIENCY IN THE VALUE-IMPACT ANALYSIS IS THE FAILURE TO CONSIDER THE INCREASED RISKS FROM ACCIDENTS OTHER THAN ATWS THAT WOULD BE IMPOSED BY CERTAIN OF THE STAFF'S

ALTERNATIVES. THIRD, THE VALUE IMPACT INFORMATION CONTAINED IN SECY-80-409 IS NEARLY IMPOSSIBLE TO FOLLOW OR UNDERSTAND. DISCUSSIONS OF VALUE IMPACT ESTIMATES ARE CONTAINED IN ENCLOSURES B, F, AND H OF THE DOCUMENT. THESE DISCUSSIONS ARE DISJOINTED AND CONFUSING, REFERRING TO ONE OR MORE DIFFERENT VOLUMES OF NUREG 0460 WITH VARIOUS DESIGNATIONS FOR THE PROPOSED FIXES AND CONTAIN UNFOUNDED AND EXCESSIVE DOLLAR VALUES FOR MAN-REM EXPOSURE. FURTHER, THE DETAILS OF THE MODIFICATIONS ASSUMED AS THE BASIS FOR THE IMPACT ESTIMATES ARE NOT STATED.

FOURTH, THE FAILURE TO RECOGNIZE THAT FEW ATWS EVENTS HAVE THE POTENTIAL OF LEADING TO SEVERE ATWS CONSEQUENCES, THAT A LIMITED SET OF SEVERE ATWS EVENTS WOULD RESULT IN MAJOR CORE DEGRADATION, AND THAT NOT ALL MAJOR CORE DEGRADATIONS EXCEED 10 CFR 100 GUIDELINES FURTHER RESULTS IN THE VALUES BEING SIGNIFICANTLY OVERESTIMATED AND IS NOT APPROPRIATE FOR VALUE IMPACT ANALYSIS.

TURNING NOW TO THE SCHEDULE PROPOSED IN SECY-80-409, IT IS SAFE TO ASSERT THAT IT IS UNACHIEVABLE AND UNJUSTIFIED IN VIEW OF THE NUMBER OF ISSUES THAT REMAIN OPEN. WE ARE BEING ASKED TO SUBMIT EVALUATION MODELS AND PLANS FOR CONFIRMATORY TESTING BY MARCH 1, 1981, AND TO PROPOSE NECESSARY MODIFICATIONS TO MEET THE CRITERIA BY JULY 1, 1981. IT IS CLEAR THAT SUCH A SCHEDULE ALLOWS NO TIME TO DO ANYTHING OTHER

THAN FALL BACK TO THE PRESCRIBED HARDWARE "FIXES" SO MUCH IN EVIDENCE IN NUREG-0460 VOLUME 4. IF CRITERIA SIMILAR TO THOSE PRESENTED IN THE PROPOSED RULE ARE ULTIMATELY DETERMINED TO BE NECESSARY, SUBSTANTIALLY MORE TIME WILL BE REQUIRED TO TEST ALTERNATIVE SOLUTIONS, PERFORM THE DETAILED ENGINEERING, AND PERFORM THE NECESSARY RELIABILITY ANALYSES TO GIVE US CONFIDENCE THAT WE ARE NOT "FIXING" OUR PLANTS IN A FASHION THAT WILL DEGRADE RATHER THAN ENHANCE SAFETY. AGAIN, WE NEED MORE EXPERIENCE WITH PRA METHODOLOGY AND IMPLEMENTATION ACQUIRED ON BASE STUDIES BEFORE WE BEGIN TO APPLY ITS RESULTS TO MAKING MODIFICATIONS TO OUR PLANTS.

THE SCHEDULE FURTHER REQUIRES THAT BWR MODIFICATIONS REQUIRED TO MEET THE ACCEPTANCE CRITERIA BE COMPLETE BY JULY 1, 1982. ON THE BASIS OF A PROPOSAL I HAVE RECEIVED FROM THE AFFECTED VENDOR IN THIS CASE, I KNOW THIS TO BE UNACHIEVABLE. WE EXPECT THE SAME TO APPLY TO THE PWR'S. FINALLY, THE SIGNIFICANT PRESSURE BOUNDARY WORK THAT MAY BE REQUIRED ON THE AFFECTED PWR'S IS TO BE COMPLETE BY JANUARY 1, 1984. SHOULD PRESSURE BOUNDARY BACKFITTING IN FACT BE REQUIRED, THERE IS A TIME FOR DOING THAT, AND IT IS DURING THE 10 YEAR IN-SERVICE INSPECTION. RESERVING ANY SUCH MODIFICATIONS FOR THAT INSPECTION AVAILABILITY WILL SUBSTANTIALLY REDUCE THE IMPACT TO THE RATEPAYER FROM NUCLEAR PLANT DOWN TIME.

OUR PROBLEMS WITH THE ACHIEVABILITY OF THE SCHEDULE ARE NOT LIMITED TO THE PLANTS WHICH NOW HAVE OR EXPECT OPERATING LICENSES BY JANUARY 1, 1984. FOR EXAMPLE, USING THE PROPOSED SCHEDULE, THE APPLICANT FOR A NUCLEAR UNIT EXPECTING TO RECEIVE AN OPERATING LICENSE IN JANUARY, 1984, SHOULD HAVE SUBMITTED PROPOSALS FOR COMPLYING WITH THE RECENTLY ANNOUNCED CRITERIA IN JANUARY, 1979.

WE SEE NO REASON FOR INCLUDING DETAILED IMPLEMENTATION SCHEDULES IN RULES AND SUGGEST THAT SUCH A PRACTICE NOT BE CONTINUED HERE. THE STAFF CERTAINLY HAS AT THEIR DISPOSAL ALTERNATIVES TO THE ESTABLISHMENT OF SUCH SCHEDULES SHORT OF INCLUDING THEM IN THE RULES.

ANOTHER MAJOR DEFICIENCY CONCERNS THE QUESTION OF THE STAFF'S LACK OF ATTENTION TO TECHNICAL DETAIL. A MAJOR PORTION OF INDUSTRY PERCEIVES THE STAFF'S "ENGINEERING JUDGEMENT" IN THIS AREA TO BE DEFICIENT. FOR EXAMPLE, THE STAFF ASSUMES THAT ALL ATWS EVENTS THAT COULD LEAD TO A CORE MELT WILL EXCEED 10 CFR 100 LIMITS. THESE ASSUMPTIONS ARE OVERLY CONSERVATIVE. THEY IGNORE THE FACT THAT EXCEEDING STRESS LEVEL C REQUIREMENTS OR EXCEEDING AN ARBITRARY TEMPERATURE LIMIT IN A BWR TORUS, DOES NOT NECESSARILY LEAD TO CORE MELT, AND CORE MELT DOES NOT NECESSARILY LEAD TO VIOLATION OF CONTAINMENT INTEGRITY OR TO EXCEEDING THE 10 CFR 100 LIMITS. THEY HAVE NOT TAKEN INTO ACCOUNT ANY OPERATOR ACTION WHICH, FOR SUCH AN EVENT, WOULD BE

A CERTAINTY. THEY OVERESTIMAT' THE NUMBER OF SIGNIFICANT EVENTS BECAUSE: (A) BELOW A CERTAIN POWER LEVEL, THE CONSEQUENCES OF AN ATWS ARE NOT SIGNIFICANT; (B) MANY ANTICIPATED TRANSIENTS WHEN COMBINED WITH A FAILURE TO SCRAM DO NOT LEAD TO BOUNDING CONSEQUENCES; (C) THE CONSEQUENCES ARE A FUNCTION OF TIME IN CYCLE; (D) NOT ALL ATWS EVENTS WILL NECESSARILY CAUSE A COMPLETE FAILURE OF THE REACTOR SHUTDOWN SYSTEM; (E) AN ATWS EVENT NEED NOT NECESSARILY CAUSE A FAILURE OF THE REACTOR CONTROL SYSTEM; AND (F) AS THE EXPERIENCE LEVEL RISES WITH ADDED YEARS OF OPERATION, THE NUMBER OF SIGNIFICANT EVENTS FALLS FOR CERTAIN CATEGORIES OF INITIATING EVENTS AS A RESULT OF THE LEARNING CURVE. THE STAFF HAS NOT TREATED IN APPROPRIATE DETAIL EVIDENCE THAT SOME OF THE MEASURES THAT HAVE BEEN RECOMMENDED TO DECREASE THE ATWS RISK MAY, IN FACT, INCREASE COMPETING RISKS, THUS, LOWERING OVERALL SAFETY.

APPROXIMATELY 20 UTILITIES REPRESENTING ABOUT 60 PLANTS HAVE PROPOSED A SOLUTION RECENTLY IN THE FORM OF A PETITION FOR RULEMAKING ON ATWS. PART 1 OF THE PETITION PROPOSES MODIFICATIONS THAT ARE STRAIGHTFORWARD AND WELL UNDERSTOOD BY THE INDUSTRY AND THE NRC STAFF. THUS, THESE MODIFICATIONS WILL NOT REQUIRE GREAT EXPENDITURES OF RESOURCES FOR TECHNICAL ANALYSIS, AND THEY CAN BE IMPLEMENTED QUICKLY. BECAUSE A SUBSTANTIAL PORTION OF THE INDUSTRY IS ALREADY WILLING TO MAKE THESE MODIFICATIONS IF THEY WILL RESOLVE THE ATWS ISSUE FOR

EXISTING PLANTS, THERE IS NOT LIKELY TO BE MUCH REGULATORY EFFORT REQUIRED TO IMPOSE THEM. MOST IMPORTANT OF ALL, THE PROPOSED MODIFICATIONS CLEARLY DECREASE THE RISK OF ATWS WHILE MINIMIZING OTHER, COMPETING RISKS.

IN ADDITION, THE PETITION PROPOSES THAT IF THE COMMISSION ELECTS TO PROPOSE ATWS MODIFICATION BEYOND THOSE IN PART 1 OF THE PETITION, THEN ALL CONCERNED WILL FIND THEMSELVES IN A MORASS OF UNANSWERED QUESTIONS DEMANDING IMMEDIATE ANSWERS AND EXCESSIVE NRC AND INDUSTRY MANPOWER REQUIREMENTS. CHIEF AMONG THESE QUESTIONS WILL BE WHETHER THE ADDITIONAL POTENTIAL MODIFICATIONS, IF IMPLEMENTED, WOULD LEAVE THE PUBLIC MORE SAFE OR LESS SAFE. THE PETITIONERS INDICATE THAT NOTHING SHORT OF AN ATWS RULEMAKING INVOLVING ADJUDICATORY PROCEDURES COULD PROVIDE THE ANSWER. THE PETITIONERS URGE THAT SUCH A RULEMAKING BE HELD IF ATWS MODIFICATIONS BEYOND THOSE IN PART 1 OF THE PETITION ARE, IN FACT, TO BE CONSIDERED NOW.

WE FEEL THAT SUCH ACTION COMING AT THIS TIME ON THIS EVENT WOULD BE UNWISE AND COUNTERPRODUCTIVE. DOING SO WOULD BE AN ATTEMPT TO PROVIDE THE ULTIMATE RESOLUTION OF ATWS IN ISOLATION FROM ALL OTHER DEGRADED CORE SCENARIOS. ONE OF THE FIRST LESSONS LEARNED FROM THREE MILE ISLAND WAS THAT NRC AND THE INDUSTRY HAD CONCENTRATED TOO MUCH ON LOW PROBABILITY EVENTS. WE SHOULD NOT FORGET THIS LESSON IN OUR EFFORTS TO IMPROVE THE SAFETY OF OUR PLANTS.

IN CONCLUSION, THE ORGANIZATIONS THAT I REPRESENT HERE TODAY HEREBY RECOMMEND THE FOLLOWING:

FIRST: THAT THE STAFF PROPOSED ACCEPTANCE CRITERIA FOR ANALYSIS OF ATWS MITIGATION CAPABILITY ALTHOUGH WELL INTENDED, ARE PREMATURE, AND SHOULD NOT BE ADOPTED AT THIS TIME.

SECOND: THAT THE COMMISSION ACCEPT THE UTILITIES' PROPOSAL CONTAINED IN PART 1 OF THE ATWS UTILITY GROUP PETITION. DOING SO WILL REDUCE THE RISK ASSOCIATED WITH ATWS BY AT LEAST 50%.

THIRD: THAT A DECISION ON WHETHER ADDITIONAL RISK REDUCTION IS APPROPRIATE AWAIT THE ESTABLISHMENT OF A SAFETY GOAL AND THE INSIGHTS TO BE GAINED IN THE NEAR FUTURE FROM THE SEVERAL ONGOING PROBABILISTIC RISK ASSESSMENT EVALUATIONS.

FOURTH: THAT AS A RESULT OF THE ABOVE, THE UNRESOLVED SAFETY ISSUE ON ATWS BE CLOSED NOW, AND ANY RESIDUAL RISK BE TREATED IN THE DEGRADED CORE RULEMAKING.