ELECTRIC POWER RESEARCH INSTITUTE

May 20, 1980

EPRI

Mr. Ashok Thadani
Reactor Systems Branch
U.S. Nuclear Regulatory Commission
Phillips Building
Washington, D.C. 20555

SUBJECT: Comments on Draft Volume 4 of NUREG-0460

Dear Mr. Thadani:

We are pleased to submit some comments on the subject Draft Report in response to the March 27 notice in the Federal Register. The enclosed Attachments and 2 contain fairly detailed analyses of specific factors inherent in Volume 4 of "Anticipated Transients Without Scram for Light Water Reactors', (NUREG-0460). These detailed analyses suggest that there are, at best, inconsistencies leading to the conclusions. At worst, there are significant flaws. We would suggest, based on our analysis, that a fairly exhaustive re-appraisal be initiated prior to dictating specifics of implementation that extend beyond those identified as Alternative 2A.

Within Attachment 1, the analyses focus upon the data base for assigning frequency and probabilities leading to ATWS. There is always technical leeway on how the extant data may be utilized and we offer a perspective which we believe is operationally sound. It is based on the manner of sorting the system data, as opposed to simply adding events. Also within Attachment 1, there is an analysis that indicates that the ATWS risk is overstated in the near term by NUREG-0460 because of the overstatement of frequency (cited above), an apparent underestimate of the scram system testing rate, a neglect of the extant "learning curve" experience, and a failure to account for future learning in estimating future risk. It is estimated that this overstatement ranges from a factor of 10 to 100.

Attachment 1, and to a lesser extent, Attachment 2 also addresses the questions of competing risk. These analyses suggest that ATWS risk can only be reasonably quantified in the context of other risk contributing events. In particular, there is an analysis, using the input data from NUREG-0460, that indicates markedly different conclusions from those arrived at in the NRC documentation.

Independently of the above statements concerning the content of the enclosed attachments and their appendices, we have serious concerns with what we perceive to be the use of technically contradictory approaches to determining the bases for declaring ATWS of importance. These concerns are briefly discussed in Attachment 3.

The results cited above are significant and seriously affect the formalities of the value-impact analyses that appear to provide the technical bases for

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the implementation suggested in NUREG-0460. Attachment 2 points out extensive inconsistencies in the analysis and suggested implementation based thereon. These inconcsistencies warrant some significant re-examination. The latter is mandatory in that implementation is both expensive and time-consuming and raises the specter of both increased risk in some cases and, at best, marginal risk reduction in many others.

We hope the above comments and the enclosures, which were prepared by the staff of the Nuclear Power Division of the Electric Power Research Institute, will be useful in your future deliberations. If there are any questions concerning their content, please contact Drs. G. S. Lellouche or I. W. Wall of my staff.

W. B. Loewenstein

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Encs.

Attachment 1

Analysis of the ATWS Assessment in NUREG 0460-IV

This analysis is an independent assessment of specific and important factors in the NUREG 0460-IV analysis that concludes that ATWS is significant. We find that there are a number of factors within the NUREG 0460-IV analysis that are conservative beyond reasonable prudence. As a result one may seriously question the conclusions of the significance of ATWS risk stated in NUREG-0460-IV and the usefulness of the proposed alternative 4A fix for PWRs. The serious factors addressed here relate to the determination of·

A. Frequency of Anticipated transients

B. Near Term (30 year) risk due to ATWS

C. Failure to consider competing risks.

The reason we address these points is that the stated rationale for requiring a specific ATWS fix is based on Value/Impact analyses of various alternative ATWS "fixes". Each Value/Impact statement has an ATWS frequency attached to it and an implied reduction in ATWS risk. In Attachment 2 the Value/Impact analysis is explicitely considered.

Conclusions

A. The anticipated transient frequency is significantly overstated because:

1. effect of initial power level is not accounted for;
2. learning effects are neglected;
3. condenser effects are not accounted for.

These considerations suggest a factor of about 5 reduction in frequency relative to that stated in NUREG-0460

B. The near term (30 year) risk of ATWS is considerably overstated because:

1. the frequency of anticipated transients is considerably overstated;
2. the testing rate for scram systems is underestimated;
3. aphysicalcommon mode events have to be postulated;
4. although future risk is estimated from plants supposedly on line for 30 years no increase in plant experience is accounted for in the risk estimation.

These conservatisms suggest a factor of 10 - 1000 overstatement of ATWS risks in NUREG-0460-IV.

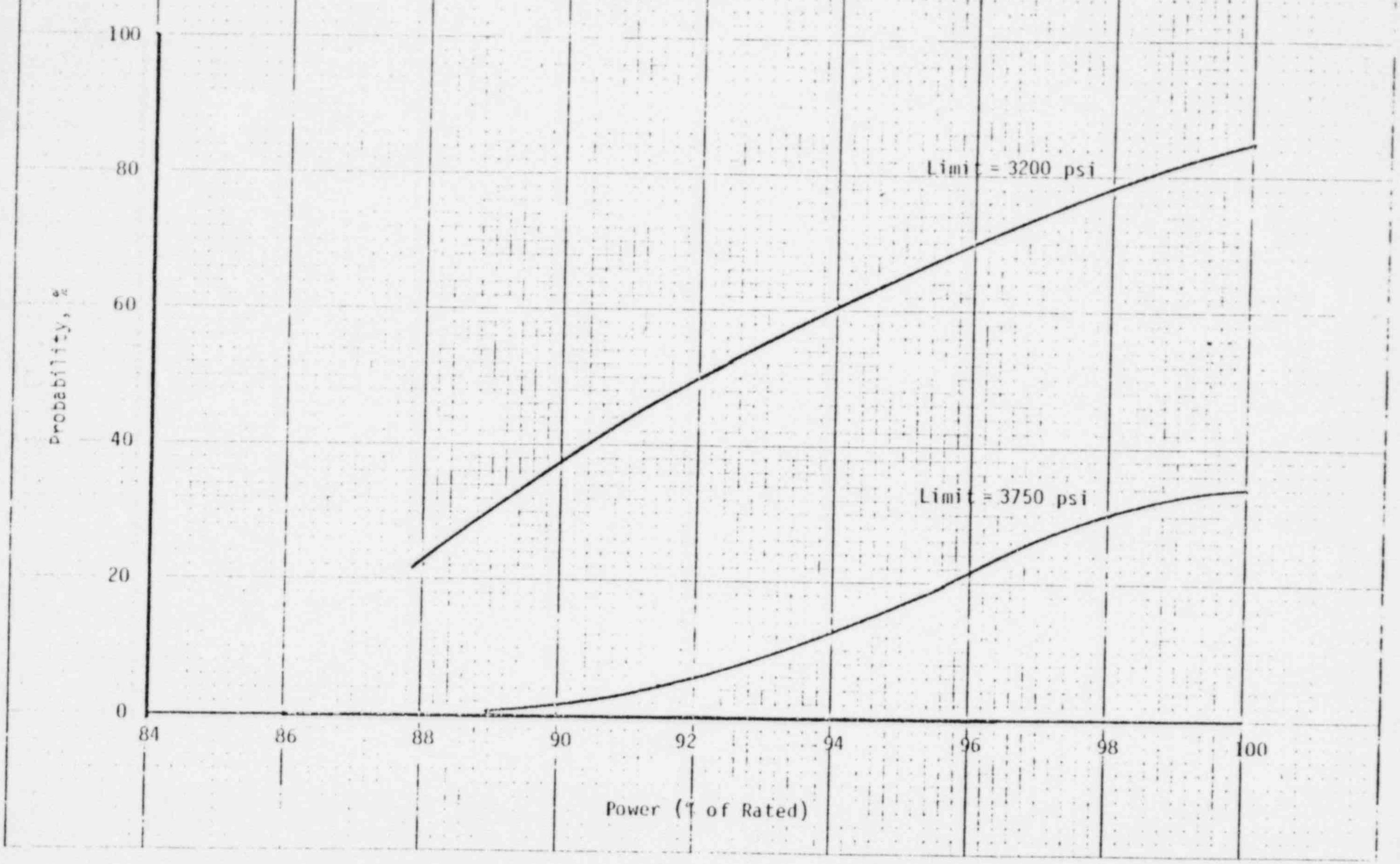


FIGURE 1: Probability of Exceeding Pressure Limit For a B & W Loss of Feedwater

2. All reactor types are the same.

Although NUREG-0460 states that 8 transients per year apply to BWRs and 5 to PWRs, it does not break down the various PWR vendors or reactor types for a vendor. Given the detailed analyses done by the vendors voer the eyars it is indeed possible to discriminate by vendor. This discrimination was done in EPRI NP-801 and the following tables (2,3) are reproduced from there. If one looks at the actual plant behavior for these events (Figure 2) one sees that the real vehavior of plants varies significantly from that assumed by the staff.

3. Learning Curve Considerations

The Analysis in NUREG-0460 does not provide for a "learning curve effect" (presumably to assure conservatism). However, it can be, and has been, shown on analytical and empirical grounds that initial and ultimate experiences differ greatly. If one does standard statistical hypothesis testing for a particular transient type, such as Turbine Trip or Generator Trip it is clear that the first year of commercial operation is different from all subsequent years with 95% confidence. This analysis was done in EPRI NP-801 using the Behren-Fischer test, but other statistical tests show the same results. Note that doing a test for the sum of all transients as the Staff did, is not meaningful since some transients (such as loss of offsite power) are time independent hence obscure learning curve effects on specific transients. Hypothosis testing shows that, as of the time of publication of EPRI NP-801, the Turbine Trip and Generator Trip transients were different at the 95% level but this was not factored into the NUREG results. This has a 20% effect on the stated frequency.

4. Condenser Capabilities

No consideration has been given to the fact that those reactors with sufficiently large condenser capability can accomodate nearly all ATWS transients without exceeding pressure or temperature limits. In BWRs, the reactor power level after a recirculation pump trip trip is down in the 25 - 30% range (NEDO 10349) for those cases where the condenser is available; hence any system with greater than 25% bypass capacity will not experience a torus overtemperature situation for those transients where the condenser is still available. Of the BWR transients listed in Table 3 only the following would impact:

MSIV (all loops)

Loss of Condenser Vacuum

Loss of Offsite Power

Loss of Aux. Power

Table 3

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

There are other transients such as Turbine Trip w/o bypass which would also impact but their frequencies are so low that they do not alter the fact that for BWRs with > 30% Bypass the ATWS transient frequency is reduced by a factor of about 3 from the stated in NUREG-0460-IV.

The same considerations apply to PWRs. After a PWR ATWS, the power level is in the 10 - 20% range. The pressure surge is caused by the power/cooling mismatch between primary and secondary. The peak pressure is strongly influenced by the availability of the condenser. Only the following transient lead to loss of the condenser:

Loss of Condenser Vacuum

Loss of feedwater (all loops)

Loss of Condenser Pumps

Loss of Station Power

There are others but they do not impact the frequency. The main initiating events which do not normally cause loss of condenser are:

Turbine Trip

Generator Trip

The result is that the PWR transient frequency can be as low as 0.4 to .7 for B & W and CE which is conisderably less than 5. We conclude that the frequencies in NUREG-0460-IV are overstated by at least a factor of 5.

B. Near Term Risk Due to ATWS

NUREG-0460 states that there are 4 chances in 7 for an ATWS during the next 30 years; this is used as in indicator that there is a clear danger to the public.

This conclusion is incorrect because it is inconsistent with the data which currently exist. The basic premise is that the ATWS frequency contains a scram failure unavailability of 3×10^{-5}/demand. Analysis of the component testing frequencies show that no portion of the scram system has a median unavailability greater than about 2 to 3×10^{-6}/demand and the total scram system is less than 5×10^{-6}/demand. The staff arrives at its number by misconstruing what is tested andhow the scram system operates (in the sense of redundancies and diversity of trip levels).

The trip level diversity rests on the various different sensors and their multiple redundancy. Several such trip levels are reached in each transient. Each trip level sensor must be assumed to fail, or else all redundant channels for each sensor must be assumed to fail. Alternatively all bistables or logic

C. Competing Risks and An Assessment of the Addition of Valves

Any safety-oriented plant modification contains within it the probability of accomplishing the goal desired and the potential for creating new or 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 it.

Examples of this trade-off are well known and some of them are:

1. The Interfacing LOCA (Event V of WASH 1400) where locking open an MOV to eliminate a single failure point for use of the LPIS increased the probability of a LOCA through the two check valves by a factor of 10.
2. Requiring the auxfeed system to actuate (post TMI) for certain events has increased the number of pressurizer emptying transients which appear to the operator as a LOCA and therefore increase the likelihood of operator misaction.
3. Closure of the blocking valves on the PORV and maintenance of the HPI has increased the number of safety valve actuations (and led to Crystal River).

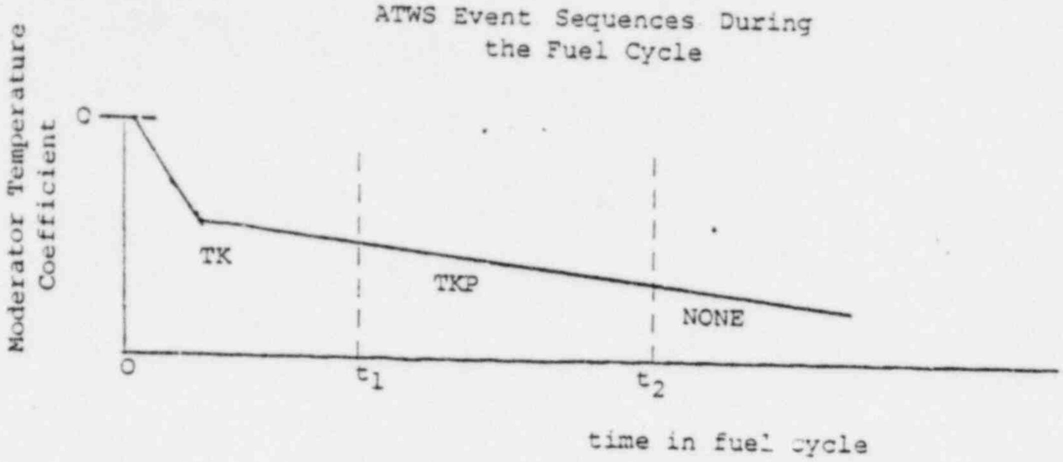
Each of these are competing risk situations where unexpected results and increased risk are obtained from a supposedly safety based modification intended to reduce risk.

In the case of ATWS the staff has suggested that increasing the number of valves on C.E. 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 (e.g.TMI-2, Crystal River).

In the following analysis, we shall deal with 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 differences.

Figure 1

ATWS Event Sequences During the Fuel Cycle



Estimate t_1 to be greater than about 30% of the cycle time and t_2 about 60%. For the very worst transients (total loss of feedwater), t_1 may be about 50% and t_2 may not apply. For our purposes we assume t_1 to be 40% and t_2 to be 80%.

COMPETING RISKS

The only competing risk we deal with here is failure of a valve to reseat. This event is denoted by Q. Clearly, for Q to occur the valve must have opened. The number of stuck open PWR valves is determined from LERs* to be 9. Using a 300 PWR reactor-year experience base, this leads to a transient frequency of 0.03/reactor year.

There are two types of event sequence where failure to reseat is significant. The first is the ATWS event itself where the sequence

TKQ

leads to a small LOCA during the entire fuel cycle. Any additional serious failure (of HPI or other necessary system) leads to a core melt. In WASH 1400 terminology these additional failures are:

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

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* A search of the NTIS tape shows ~~10~~ 9 stuck-open valves at various power levels. Using 300 reactor years of PWR experience yields 0.03 events/year. See App. 2.

Note that these events occur throughout the fuel cycle with about equal likelihood but can be expected to increase in the future since new operational requirements* have resulted in an increase in T* (a different competing risk situation). Returning to WASH 1400, the data indicated that

$D = 9 \times 10^{-3}$/demand

$F = 1 \times 10^{-4}$/demand

$G = 6 \times 10^{-3}$/demand

$H = 6 \times 10^{-3}$/demand

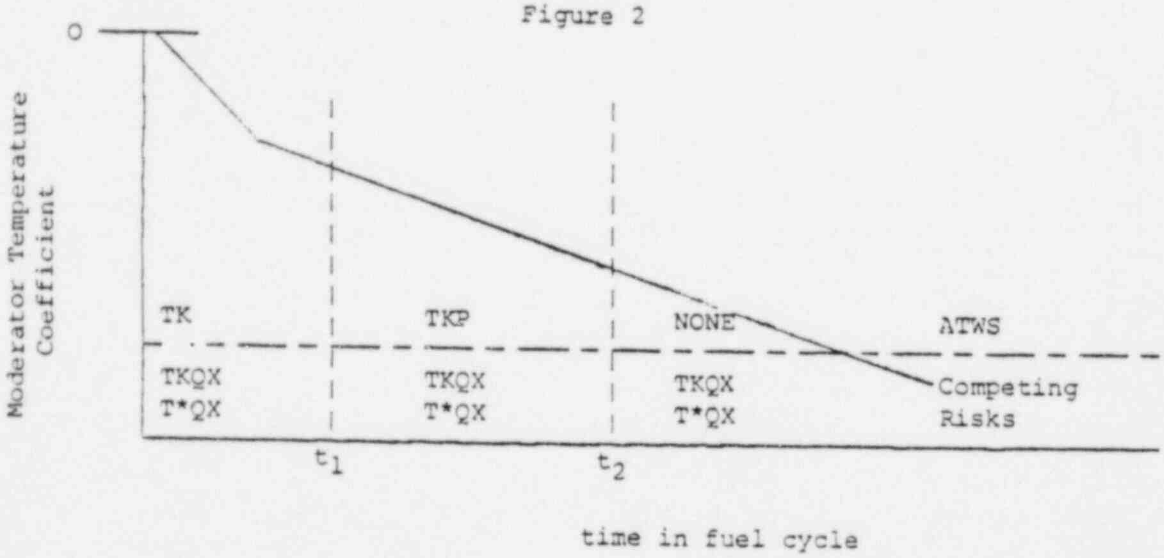
$C = 2 \times 10^{-3}$/demand

hence that

$$(T^*Q)\ (D + F + G + H + C) \equiv T^*QX = 7 \times 10^{-4}$$

we may now establish a structure for the ATWS competing risk situation based on the following figure:

Figure 2



In this structure the competing risks are independent of the fuel cycle. In order to determine the effect of adding valves one must compare

$$TK\ (t_1) + TKP\ (t_2 - t_1)$$

versus

$$TKQX + T^*QX$$

We maximize the ATWS risk reduction by assuming $t_1 \approx 1$ in which case we can compare

$$TK \text{ vs. } TKQX + T^*QX$$

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* Arising out of TMI-2 considerations relative to closing blocking valves

Release as Equivelent Iodine-131

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

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

NUREG 0460 states (App. X-7,Vol.2) that there is some possibility that an ATWS could fall into PWR-2 or PWR-3 but does not quantify this statement. It concludes the most likely failure mode is PWR-7. If we assume that all ATWS fall into the worse PWR-5 (rather than 7 as WASH 1400 indicates), and assume that all ATWS are TK but account for $t_1 \approx .4$ then the ratio of risk from stuck open valve sequences in PWR-3 to that from ATWS sequences in PWR-5 would be:

$$\frac{\left(T^*Q(D+H)-\alpha+T^*Q(F+C+G)-\delta+TKQ-\alpha\right)C_3}{(TK-\beta)\ t_1 C_5} \approx \frac{.03(.015\times.99+.006\times.01)+1.6\times10^{-4}\times.01^2)}{1.6\times10^{-4}\times.4\times.004} \times 20$$

$$\approx \frac{4.5\times10^{-4}}{2.6\times10^{-7}} \times 20$$

$$\approx 34000$$

This analysis points out first that ATWS risk for PWRs is trivially small compared to other reactor risks and second the likelihood of increased risk due to addition of valves. Any additional valves installed are not likely to have set points at the lowest pressure levels but it is inconceivable that such valve additions would not increase T*Q by 1%. Since the T*Q risk outweighs ATWS by 34000 the net change in risk due to eliminating ATWS by adding valves must be unfavorable. If a 3% increase in T*Q is reasonable, the public risk is increased by 1000 times the original ATWS risk. This is illustrated in Table 4.

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

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

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 Valves 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

Consider a staggered testing of a two unit series system. The units have failure rates of λ_1 and λ_2 respectively, and we assume the parts behave exponentially. Thus, were the system to be tested as a whole every T units of time the mean unavailability would be (let $\lambda \equiv \lambda_1 + \lambda_2$)

$$U = \frac{1}{\lambda T}(1 - e^{-\lambda T}) - e^{-\lambda T}$$

and if λT is small

$$U = \lambda T/2$$

If the system is tested in a staggered fashion the result is considerably more complex but can be shown to be no different from a single test (to first order). The probability that the system is up between (o,T/2) is

$$Pr_{12} = \int_o^{T/2} f_1\ (t)dt \int_o^{T/2} f_2\ (t + T/2)dt$$

and the pdf for the system being up at anytime between (0,T/2) is

$$f_{12}(t) = \tfrac{1}{2}\left(\lambda_1 e^{\lambda_1 t/2} - (\lambda_1 - \lambda_2)\ e^{(\lambda_1 - \lambda_2)t/2} + \lambda_2 e^{-\lambda_2 t/2}\right).$$

If f_{21} (t) is a permutation of indicies on $f_{12}(t)$ then it is easy to show that the system unavailability in a staggered test procedure is:

$$U = \frac{1}{T}\left[\int_o^{T/2} tf_{12}(t)\ dt + \int_o^{T/2} tf_{21}\ (t)\ dt\right]$$

$$= (\lambda_1 + \lambda_2)\ T/2$$

Hence there is no mathematical difference (to first order) between whole and staggered testing. There is still a question of the disparate rates of testing and the question of redundencies in the various parts of the system (particularly at the breaker end). It would be possible to set up a detailed mathematical model of the scram system including redundencies at all levels and examine the failure likelihood. We have not done this. Instead we approximate the system by its channel testing rate and do not include the effect of redundancies in the calculation. These results are found in the following table:

Some questions may be raised concerning the breaker failure rates and whether breakers dominate the RPS unavailability. The following table shows this is not true.

ASSUMMING BREAKERS DOMINATE SCRAM FAILURE FOR PWR's

No. of Breaker Failures	$\approx$ 20
Reactor Years of Experience	$\approx$ 300

		50%	95%
Failure Rate/year		6.7×10^{-2}	9.7×10^{-2}
Single Breaker Unavailability/Demand			
24 tests/year		1.5×10^{-3}	2×10^{-3}
48 tests/year		7.5×10^{-4}	1×10^{-3}
Unavailability of all Breakers/Demand			
24 tests/year	1/2	2.2×10^{-6}	3.9×10^{-6}
	2/4	$<< 10^{-6}$	$<< 10^{-6}$
48 tests/year	1/2	5.6×10^{-7}	1.1×10^{-6}
	2/4	$<< 10^{-6}$	$<< 10^{-6}$

Conclusion is that Breakers do not dominate RPS unavailability.

Finally we note that we have not considered common mode failures (except KAHL) in these analyses. It would appear that any statement is possible concerning common mode failure behavior, sufficiently so to vitiate any statement about anything. We propose that for determining a Value/Impact the postulation of

Appendix 2

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 2, 1978, PORV
7. Ft. Calhoun, 1979, 2 PORV
8. TMI-2, 1979, PORV
9. Crystal River, 1980, Safety Valve

ATTACHMENT 2

OBSERVATIONS OF VALUE-IMPACT ANALYSIS IN NUREG-0460

SUMMARY & RECOMMENDATIONS

In Executive Order 12044, President Carter required federal agencies to analyze new and existing regulations in order to determine the need (value) and economic consequences (impact). Although NRC, as an independent regulatory agency, is not covered by this Executive Order, it does have a policy requiring value-impact analyses for each significant change in regulatory requirements.[a] Based upon discussion in this attachment, NRC's value-impact analysis contained in NUREG-0460 appears to have some severe technical flaws and does not technically support the proposed ATWS requirements. The major flaws are discussed in the following paragraphs and the discussion is amplified in succeeding sections.

First, values and impacts for Alternatives 3A and 4A are stated in terms of their effect upon the existing situation. A more correct statement would be their incremental values and impacts. For example, if Alternative 2A has been implemented, it becomes the new basis and the value and impact of Alternative 3A is reduced thereby. After correcting the presentation format, it will be shown that the values of most of NRC's proposed requirements under Alternatives 3A and 4A are zero. It should be emphasized that this nullification of value occurs without any change in NRC's technical or economic assumptions.

Second, the values are generally overstated. The variation in the consequences of potential core melting is ignored and, for existing PWRs other than the one assessed in WASH-1400, NRC's estimate of public risk is shown to be high by at least a factor of 10. Further, competing risks are not substantively addressed. As shown in Attachment 1, competing risks due to implementation of some of NRC's requirements will certainly reduce the corresponding estimated values and will probably negate them, i.e. the public risk will be greater after NRC's 'fix' than beforehand. It will be shown that a more careful value analysis would not support Alternatives 3A and 4A for PWRs. The comment period was insufficient to fully address BWRs.

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[a] SECY-77-388, SECY-79-8 and NRR Office Letter No. 16

completed for a large fraction of operating or pending plants under the sponsorship of Office of Nuclear Regulatory Research, EPRI, and various utilities. By assigning sufficient priority, NRC could review these assessments for realism and consistency such that a comprehensive overview could be available within 18 months. These risk assessments should identify the dominant accident sequences including uncertainty bands on their probability assignments.

2. Use the risk profiles generated under Item 2 to re-assess the effect on public risk of the proposed requirements under Alternatives 3A and 4A, including potential adverse competing risks. In addition, examine potential risk reduction for other safety concerns, e.g. station black-out. Determine the value-impact ratio and fractional risk reduction for each safety concern including ATWS.

3. Utilize the results of Item 3 to rank each potential safety improvement according to its value-impact ratio and fractional risk reduction.

4. Continue development of a policy on 'how safe is safe enough?' It is essential for industry and NRC to have consistent guidance on a numerical bound for acceptable risk. The current NRC schedule calls for a draft statement to be submitted to the Commission by December, 1980.

The above systematic approach to reactor safety should guide industry and NRC in allocating resources to all safety concerns, including ATWS.

TABLE 2.

CORRECTED SUMMARY OF VALUES CALCULATED BY NRC (1980 $M)

	Alt. 1	Alt. 2A*	Alt. 3A	Alt. 4A
		OPERATING PLANTS		
B&W/CE	0	2.2 - 8.7	2.2 - 8.7	4.3 - 17.4
W	0	4.3 - 17.4	4.3 - 17.4	4.3 - 17.4
GE	0	13.9 - 34.5	24.0 - 60.0	25.3 - 62.8
		PLANTS UNDER CONSTRUCTION		
B&W/CE	0	2.3 - 8.6	2.3 - 8.6	4.6 - 17.3
W	0	4.6 - 17.3	4.6 - 17.3	4.6 - 17.3
GE	0	14.1 - 34.0	24.2 - 58.8	25.6 - 61.9

*Values for Alt. 2A are inferred from Table 1.

The above inconsistency proved troublesome for us to identify and correct since insufficient detail was made available about the value assessments in Volumes 2 and 4. These insufficiencies prevented us from reproducing some of the calculations in Volumes 2 and 4. For example, only aggregate values are stated in Volume 4 and, with the numerous changes in assumptions as work proceeded from Volume 2 through Volume 3 to Volume 4, it was impossible to determine which were the dominant contributions to value. Further, some assumption changes were not identified. We greatly appreciate the cooperation of NRC staff in making the revised Table 2 and its backup calculations available during the comment period. Without this information, our comments on draft Volume 4 would have been less incisive.

PRIOR TO PUBLICATION OF VOLUME 4, THE ISSUANCE OF ORDERS AND THE INITIATION OF RULEMAKING, IT IS RECOMMENDED THAT ALL CALCULATIONAL DETAILS BE MADE AVAILABLE.

OVERSTATEMENT OF RISK DEDUCTION ACHIEVABLE FROM ATWS REQUIREMENTS

The reduction in public risk from potential ATWS accidents achievable from NRC's requirements is overstated for at least two reasons. First, the variation in the consequences of potential core melting accidents is ignored. All potential ATWS accidents are assumed to lead to core melting and risk reduction is gauged solely by the smaller probability of such a consequence. Second, the additional hardware being required may increase the probability and/or consequences of other accident sequences (competing risks discussed in Attachment 1).

The Reactor Safety Study assigned each accident sequence to a Release Category primarily according to the magnitude of its atmospheric release magnitude. As shown in Table 4 below, there is a wide variation in public consequences from the different release categories. In the Reactor Safety Study, the assignment of accident sequences to different release categories

TABLE 4.

EXPECTED POPULATION DOSES FROM RELEASE CATEGORIES IN REACTOR SAFETY STUDY (PERSON-REM PER REACTOR-YEAR)*

PWR-1	17	BWR-1	24
PWR-2	123	BWR-2	126
PWR-3	105	BWR-3	226
PWR-4	5	BWR-4	3
PWR-5	2		380
PWR-6	3		
PWR-7	<0.01		
	260		

*The population doses are calculated from data presented in Tables 7 and 10 of NUREG-0340. The average population dose is 300 PR/R-Y which is slightly higher than the 280 PR/R-Y stated in Volume 2 of NUREG-0460 reflecting correction of a programming error.

cladding temperature, calculated by customary licensing methodology, exceeded 2200° F. Experiments at LOFT and elsewhere have shown the licensing methodology to be conservative. Further, experience at Three Mile Island demonstrated that severe cladding degradation is not synonymous with 100% core melting. This later work and experience suggests that the public risk from potential ATWS accidents in PWR's is even smaller than suggested above. It should also be noted that other risk assessments[a] of PWR's have neither substantively reassigned ATWS sequences to other release categories nor found ATWS sequences to be dominant contributors to public risk.

The situation with respect to BWR's is rather different. In the Reactor Safety Study assessment, the ATWS sequence was a dominant contributor to 'core melting' probability and was assigned to BWR-3 which was also a dominant contributor to public risk. NRC increased the probability of ATWS by a factor of 20, in which case ATWS would contribute about 2260 out of a total of 2526 PR/R-Y or 89% of the public risk. Thus, NRC's assignment of 1000 to 3000 PR/R-Y to BWR ATWS sequences is consistent with its assumption on ATWS probability.

It is instructive to modify the values for B&W, CE and W plants stated in Table 3 to show the effect of a more realistic estimate of reduced radiological risk, viz. 5 to 50 PR/R-Y compared to 50 to 500 PR/R-Y assumed by NRC.[b] In Table 5 below, the radiological portion of values has been reduced ten-fold.

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[a] Asselin, S. V., etal "Dominant Accident Sequences for an Ice Condenser PWR Plant" Trans. ANS, 30 p. 355, November 1978

The German Risk Study, Verlag TÜV Theinland

Garcia, A. A. "Crystal River-3 Safety Study" Preliminary S.A.I. report to NRC, 9 May, 1980.

[b] In Volumes 3 and 4 of NUREG-0460, NRC increased the ATWS probability at existing plants to $8x10^{-5}$/R-Y. There was a corresponding increase (undocumented) in public risk to 80 to 800 PR/R-Y. For simplicity, the analysis reduces NRC's estimate to 8 to 80 PR/R-Y.

TABLE 6.

MODIFIED INCREMENTAL VALUES AND IMPACTS TO SHOW EFFECT OF COMPETING RISKS FOR B&W/CE PLANTS (1980 $M)

	Alt. 2A		Alt. 3A		Alt. 4A	
	Impacts	ues	Impacts	Values	Impacts	Values
			OPERATING PLANTS			
B&W/CE	0.7	2.2 - 8.7	2.0	0	1.7	NEGATIVE
			PLANTS UNDER CONSTRUCTION			
B&W/CE	0.5	2.3 - 8.6	1.4	0	1.0	NEGATIVE

Examination of Table 6 shows that, without changing NRC's technical and economic assuptions but including competing risks, only Alternative 2A is justified for B&W/CE plants. It should be acknowledged that mitigative measures other than additional safety/relief valves might still have a favorable value/impact ratio.

The above analysis of competing risks was limited to Alternative 4A requirements for B&W/CE plants. Due to the limited period for comments on Volume 4, time was unavailable to address the potential competing risks raised by NRC requirements for other plants. Such competing risks almost certainly exist.[a]

PRIOR TO PUBLICATION OF VOLUME 4, THE ISSUANCE OF ORDERS AND THE INITIATION OF RULEMAKING, IT IS RECOMMENDED THAT NRC REVISES ITS VALUE ANALYSIS TO REALISTICALLY ACCOUNT FOR THE VARIATION IN THE CONSEQUENCES OF POTENTIAL CORE MELTING ACCIDENTS AND TO COMPREHENSIVELY ADDRESS POTENTIAL COMPETING RISKS.

[a] L. H. Heider Letter to Dr. Milton S. Plesset, Chairman, ACRS, 7 April 1980.

Throughout NRC's value-impact analysis for ATWS, there are numerous conservative technical and economic assumptions at each stage[a]; many of them are acknowledged by NRC. These conservatisms bias the value-impact analysis to support NRC's proposed requirements for ATWS but, in a global sense, is the public interest being served? Apart from the competing risks addressed elsewhere, the disadvantages of NRC's proposals are less tangible. Since NRC's and industry's (and United States) resources are finite, manpower and monies applied to ATWS inevitably detract resources from other applications which may be more productive. Professor Okrent eloquently expressed this concern:[b]

> "Society uses the word safe in a vague and inconsistent fashion. Efforts to reduce risk are not necessarily made in the most cost-effective way. Our priorities should be reevaluated. In view of their statistically smaller contribution to societal risk, major accidents may be receiving proportionately too much emphasis compared to other sources of risk ... Society's resources are limited ... Above a particular level, expenditure of resources on additional programs to reduce risks to health and safety may be counterproductive because of adverse economic and political effects."

Several independent observers[c] have suggested that, pre-TMI, NRC's disproportionate emphasis on large-LOCA precluded sufficient attention to small-LOCAs, transients and human factors.

In order to gauge the magnitude and effect of NRC's conservative assumptions, their estimated values have been recalculated with (a) more realistic economic parameters and (b) EPRI's more realistic estimates of

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[a]Although not apparent in this application, introducing separate 'conservatisms' at many stages can also be misleading. For example, in the calculation of seismic stresses, a 'conservative' calculation of soil-structure interaction may bias the frequency content which is non-conservative at a later stage. A safer practice is the use of realistic calculations throughout with one conservative safety factor at the final stage.

[b]Okrent, D. 'Comment on Societal Risk' Science 208 25 April 1980.

[c]Risk Assessment Review Group (1978), Kemeny Commission Staff (1979) especially the supplemental view of Commissioner Pigford, and Rogovin report (1980).

TABLE 7.

MODIFIED INCREMENTAL VALUES AND IMPACTS BY USING REALISTIC ECONOMIC PARAMETERS (1980 $M)

	Alt. 2A		Alt. 3A		Alt. 4A	
	Impacts	Values	Impacts	Values	Impacts	Values
			OPERATING PLANTS			
B&W/CE	0.7	1.3 - 3.1	2.0	0	1.7	1.3 - 3.1
W	0.5	2.5 - 6.2	1.2	0	0.8	0
GE	1.0	5.6 - 11.8	2.5	4.1 - 8.6	9.5	0.5 - 1.1
			PLANTS UNDER CONSTRUCTION			
B&W/CE	0.5	1.3 - 3.1	1.4	0	1.0	1.3 - 3.1
W	0.4	2.5 - 6.2	0.8	0	0.6	0
GE	1.0	5.6 - 11.7	2.2	4.0 - 8.5	7.6	0.5 - 1.1

Attachment 1 discusses NRC's conservative assumptions for the frequency of pertinent transients and for unavailability of reactor protection system. Based upon a more realistic analysis of historical data from operating plants, EPRI has conservatively estimated the probability of severe consequences from ATWS in existing plants to be about $2x10^{-6}$, less than 10^{-6} and $1.8x10^{-5}$ per reactor-year for B&W/CE, W, and GE plants respectively. By approximately modifying these probabilities to account for the effect of NRC's proposed requirements, Table 1 is restated below as Table 8.

TABLE 8.

EPRI's APPROXIMATE ESTIMATES OF FREQUENCY OF SEVERE CONSEQUENCES FROM ATWS EVENTS IN LWR's ($x10^6$/REACTOR-YEAR)

	Existing	Alt. 2A	Alt. 3A	Alt. 4A
B&W/CE	2	1	<1	<1
W	<1	<1	<1	<1
GE	18	8	1	<1

If these frequencies are now translated to incremental values by using NRC's economic parameters, the following Table 9 is obtained for operating plants.

A MORE COMPREHENSIVE APPROACH

While NRC's value-impact analysis is required as a matter of policy and is certainly a major figure-of-merit, it is an insufficient basis for a decision to allocate resources to 'fixing' ATWS.

First, it lacks perspective since it considers the ATWS issue in isolation. The potential reductions in public risk by 'fixing' ATWS are very small fractions of the overall risk imposed by the nuclear power plant. Table 1 stated NRC's estimates of the frequency of severe consequences from ATWS events both for existing designs and after proposed 'fixes'. However, the probability of severe consequences due to non-ATWS events is assumed by NRC to be unchanged by these fixes. Based upon data presented in Appendix F of Volume 3 (NUREG-0460), NRC estimates this non-ATWS probability to be 2.9×10^{-5}/r-y[a]. By adding this probability to those stated in Table 1, the impact of NRC's requirements for ATWS upon the overall probability of severe consequences is obtained and is stated in Table 10.

TABLE 10.

NRC'S ESTIMATE OF TOTAL FREQUENCY OF SEVERE CONSEQUENCES IN LWRs ($X10^6$/REACTOR-YEAR)

	EXISTING PLANTS	PROPOSED ATWS REQUIREMENTS		
		2A	3A	4A
B&W/CE	110	70	70	30
W	110/30	30	30	30
GE	230	120	40	30

It is evident by comparing Tables 1 and 10 that the fractional risk reductions due to the proposed ATWS requirements are substantially less than the reductions for ATWS alone. For example, by expending $1.7 million on a B&W/CE plant (incremental impact for Alternative 4A), NRC judges that the ATWS probability is reduced from 4×10^{-5} to 10^{-6}/r-y (∿100% decrease) but the overall probability is only reduced from 7×10^{-5} to 3×10^{-5} (∿60% decrease). Furthermore, incorporation of competing risks (Attachment 1 and earlier

[a] This probability will vary from plant to plant but the argument is still valid.

approach to improving reactor safety should also estimate the value/impact ratios and the fractional risk reductions potentially achievable by allocating resources to other dominant accident sequences as well as ATWS. For example, recent NRC requirements to 'fix' check valve (V) and auxiliary feedwater systems almost certainly had higher value/impact ratios and greater fractional risk reductions than those being claimed for ATWS. There may be other issues with greater merit.

Third, NRC's proposed requirements are predicated upon the perceived need to achieve a probability of 10^{-6}/reactor-year for severe consequences due to ATWS-initiated accidents (Section 5, Volume 1, NUREG-0460)[a]. This 'goal' was set by the NRC staff with the perception (page 35, Volume 1, NUREG-0460) that future nuclear power plants would be safer but lacked the benefit of the broader perspective of risk within American society which should indicate any need for greater safety. It is our understanding (Meeting with W. E. Vesely, Probabilistic Analysis Staff, 18 March 1980) that NRC plans to submit recommendations on numerical risk criteria in January 1981. It would seem reasonable to reassess the aforementioned ATWS goal at the time of this submittal. EPRI expects to contribute to the formulation of quantitative risk criteria in another forum.

Based upon the above observations, a more comprehensive approach to improving reactor safety would include the following tasks:

1. Generate overall probabilistic risk assessments for a representative spectrum of nuclear power plants. The following probabilistic risk assessments either have been, or will be, completed by the end of calendar 1981 under the sponsorship of NRC's Probabilistic Analysis Staff, EPRI or utilities:

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[a] The numerical safety objective was rejected in the later Volume 3 although the probability target remained unchanged. Since issuance of that volume, the TMI-2 Lessons Learned Task Force (Section 4.1, NUREG-0585) and the Rogovin Report (Recommendation 8, NUREG/CR-1250, VOL 1) have strongly reccommended a substantive quantitative risk objective for nuclear power plants. Considering these strong recommendations, it is somewhat surprising that draft Volume 4 (NUREG-0460) merely reiterates the earlier statements in Volume 3.

ATTACHMENT 3

CONCERNS WITH METHODOLOGY

ATWS is not a simple phenonmenon. It is a question concerning the response of a system to a number of possible transient initiators when the exact state of the system initial conditions are not precisely known; yet, the response of the system strongly depends on these initial conditions. Further, the consequences depend not only on this initial state but precisely which transient is considered.

The approach taken by NUREG-0460 has superficially exhibited some of the variability of the consequences of initial conditions and of some of the various initiators. However, such analyses were limited to small variations in initial power levels, did not account for variation in condenser capacity, and ultimately did not account for the fact that the various transient initiators have widely differing frequencies.

A conclusion that ATWS is important because there are several event initiators averaging, according to NUREG-0460, 5 events per year for all power levels and the worst transient yields unacceptable consequences at the worst power level is technically insufficient, at best. There are mathematically well based methods for doing a sound analysis and they should have been used. One such method is described below.

Let $f_i(x_j)$ be the frequency of the i^{th} transient and $p_i(x_j)$ the peak pressure depending on the initial conditions x_j, then the following mathematically meaningful results could have been obtained:

1. Expected peak pressure for ATWS

$$\bar{p} = \frac{\sum_{i,j} f_i(x_j)\ p_i(x_j)}{\sum_{i,j} f_i(x_j)}$$

2. The probability of exceeding a given pressure.
3. The expected peak pressure and probability distribution as a function of initial power level, or condensor capacity, etc.

Such results would have been technically defensible and less likely to lead to the degree of controversy that has surrounded ATWS. The EPRI study in this area (EPRI NP-1090) shows seriously different perceptions than are implied by the statements in NUREG-0460, indeed, they are sufficient to cast doubt on several conclusions reached in NUREG-0460.