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A. Thadani
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Note to: S. Hanauer

Attached are the requested definitions of possible ATWS fixes and a summary of the arguments for and against each fix. Four fixes representative of a range of possible requirements from no additional design changes to adding consequence mitigating systems were studied. Some arguments required more knowledge of the details of a fix than is currently available; however, the group filled these gaps as best they could. Obviously all of these fixes would require analyses to determine details of the design.

Some of the arguments for or against fixes are dependent on the desired safety goal (e.g. a less restrictive goal justifies fewer design changes). Similarly, acceptance criteria (e.g., system pressure) and reliability assessment will affect the judgments.

J. Murphy

Attachment:
As stated

Z - Current vendor designs are acceptable without modification

Pro

1. Available data do not contradict claims for high reliability for the reactor protection system. Thus, ATWS may have a sufficiently low probability that it need not be considered.

2. Operating plants, because of their small number, need not be modified since they do not significantly contribute to the overall risk from a 1000 reactor industry.

3. Additional complexity introduced by an ATWS fix may increase the likelihood of system failure during both normal operation and accident conditions.

4. Economic advantage.
Funds required to achieve ATWS modifications may be better utilized to resolve other safety problems.

Con

1. Industry probability analyses are inadequate to demonstrate that the unavailability of the reactivity shutdown system is sufficiently low. Transient frequencies may be higher than presented in WASH-1270; thus, leading to an increased probability of ATWS.

2. It is desirable to maintain approximately the same level of safety at all plants.

3. If properly designed, modifications resulting from an ATWS fix should not significantly increase the likelihood of system failure during non-ATWS conditions and may improve it.

4. May ignore cost-effective fixes.

5. Inconsistent with present NRC positions.

M - Improve reliability of RPS by requiring independence and diversity from sensors to scram relays. No other modifications required.

Pro

1. Industry probability analyses are inadequate to demonstrate that the unavailability of the reactivity shutdown system is sufficiently low; and, therefore, independent backup should be provided.

2. Improves reliability of the reactor protection system (sensors to scram relays) and may reduce the likelihood of ATWS to an acceptable level.

3. If properly designed, modifications resulting from an ATWS fix should not significantly increase the likelihood of system failure during non-ATWS conditions and may improve it.

4. Cost-effective modification.

Con

1. Modification unnecessary in light of "pro" arguments presented in Z above.

2. Proposed modification does not affect the likelihood of failure of the control rod drive system. The likelihood of such failures could dominate the likelihood of failure to scram.

3. It may be difficult to demonstrate the reliability independence and diversity in the reactor protection system design as suggested by this fix.

4. May ignore other possible cost-effective fixes.

5. Inconsistent with present NRC positions.

V - Improve reliability of RPS by providing independence and diversity from sensors to scram relays. Require recirculation pump trip for BWR. Require diverse auxiliary feedwater initiation and turbine trips for PWRs.

Pro

1. Industry probability analyses are inadequate to demonstrate that the unavailability of the reactivity shutdown system is sufficiently low. The proposed fix improves reliability of the reactor protection system (sensors to scram relays) and may reduce the likelihood of ATWS to an acceptable level as well as mitigate the consequences of such an event.

2. Recirculation pump trip is being implemented on operating reactors. Recirculation pump trip with manual initiation of current SLCS and HPIS designs may make the consequences of ATWS acceptable. The BWR analyzed in WASH-1400 incorporated recirculation pump trip with manual SLCS operation. ATWS was found to contribute approximately 30% of the overall probability of core melt. Installation of a recirculation pump trip reduces the power spike associated with a turbine trip.

Con

1. Proposed modification does not affect the likelihood of failure in the control rod drive system. The likelihood of such failures could dominate the likelihood of failure to scram.

2. Analysis subsequent to WASH-1400 indicates that pump trip without fast acting SLCS of higher capacity than in present designs may not prevent core melt. Recirculation pump trip installation aggravates the reactor coolant system pressure resulting from a turbine trip. Recirculation pump trip places the plant in a natural circulation mode, a less stable hydrodynamic state. Electrical braking of the recirculation pump during a LOCA will not be available if the pump trips due to a turbine trips, a feature not presently incorporated in the design. This could lead to missile generation due to motor failure.

V - Improve reliability of RPS by providing independence and diversity from sensors to scram relays. Require recirculation pump trip for BWR. Require diverse auxiliary feedwater initiation and turbine trips for PWRs. (continued)

Pro

3. Installation of diverse auxiliary feedwater system initiation and diverse turbine trip at certain PWRs may mitigate the consequences of ATWS. The PWR analyzed in WASH-1400 contributed less than 10% to the overall probability of core melt. Installation of diverse auxiliary feedwater system initiation may improve the availability of auxiliary feed-water for non-ATWS events.

Con

3. Without more relief capacity, turbine trip and auxiliary feedwater in plants not studied in WASH-1400 may not prevent core melt.

4. It is difficult to demonstrate reliability, independence and diversity in the modified reactor protection system design.

5. Modification unnecessary in light of "pro" arguments presented in Z or M above.

I - Improve reliability of RPS by providing independence and diversity from sensors to scram relays. Require recirculation pump trips and automated improved standby liquid control system (SLCS) for BWRs. Require diverse feedwater and turbine trips and an improved boron addition system for PWRs. For certain PWRs, require additional relief valves. Perform analyses assuming no additional failures.

Pro	Con
1. Industry probability analyses are inadequate to demonstrate that the unavailability of the reactor shutdown system is sufficiently low. The proposed fix improves the reliability of the reactor protection system and provides a fast acting boron addition system to backup the control rods. Likelihood and consequences of ATWS are sufficiently low that additional failures need not be considered.	1. Modification unnecessary in light of "pro" arguments in either Z, M, or V, above.
2. Addition of relief valves may prevent overpressure failure at those PWRs where analysis indicates diversity in turbine trip and auxiliary feedwater system initiation does not suffice. Addition of increased relief capacity could provide additional protection for non-ATWS events.	2. Installation of extra relief valves leads to increased probability of inadvertent blowdown. Increasing the number of relief valves per plant would increase the likelihood that one valve does not reclose after all valves open.
3. Faster addition of boron at a PWR or automatic in a BWR will provide a means to terminate transients in the long term. Faster addition of boron will improve the capability for water addition and negative reactivity insertion for non-ATWS accidents.	3. Boron addition at a PWR will not affect the early portion of the pressure transient and may not be cost effective. Addition of means of fast injection of borated water at high pressure at a PWR may increase the likelihood of overpressurization events initiated at low temperature. Some improved boron addition designs in a PWR could lead to more severe boron dilution events.

I - Improve reliability of RPS by providing independence and diversity from sensors to scram relays. Require recirculation pump trips and automated improved standby liquid control system (SLCS) for BWRs. Require diverse feedwater and turbine trips and an improved boron addition system for PWRs. For certain PWRs, require additional relief valves. Perform analyses assuming no additional failures. (continued)

Pro	Con
4. Installation of a recirculation pump trip reduces the power spike associated with a turbine trip.	4. Recirculation pump trip installation aggravates the reactor coolant system pressure resulting from a turbine trip. Recirculation pump trip places the plant in a natural circulation mode, a less stable hydrodynamic state.
5. Installation of diverse auxiliary feedwater system initiation may improve the availability of auxiliary feedwater for non-ATWS events.	5. Electrical braking of the recirculation pump during a LOCA will not be available if the pump trips due to a turbine trips, a feature not presently incorporated in the design. This could lead to missile generation due to motor failure.

III. Probability Estimation for Scram Systems

A. Data

1. Systems Data

EPRI gives the following summary of operating experience through 1975:

Units	Accumulated Reactor Years	Total Scram Demands* (Estimated)
U.S. Comm. Power	228	Total = 39,212
Army	57	
N.S. Savannah	10	
Foreign Comm. Power	673	
Navy	1252	75,120
Total	2220†	114,332

*Includes tests and partial tests.

†WASH-1270 used 1627 Reactor Years.

Two known failures have occurred, both early in reactor development, and there is some question of whether one or both should be "counted" in drawing inferences from these data. There are also questions as to whether all the estimated scram demands test the system to the same extent that an anticipated transient would and whether it's reasonable to assume no navy failures.

2. Component Data

Attachment B, page 10, lists BWR rod failures identified by EPRI from the Nuclear Safety Information Center. Among BWR's there have been six incidents of less than full insertion of one or more rods. The number of affected rods ranged from 1 to 96. Three of these rods failed to insert at least to position 02 and were counted by EPRI as failures.

For PWR's, EPRI considers only failures at newer reactors which have a different control rod design from older reactors. For the new design, there have been 2 failure incidents, each involving one rod failure.

It appears that the only failure data included in the EPRI summaries are mechanical failures. Calibration errors and failures of the electronic components of scram systems are not included. This is an area where more information is needed.

3. System and component failures have happened. Thus, there is more to work with than the no-alligators-in-the control-room situation.

B. Methods

1. There are two approaches which have been used to obtain estimates of the probability of scram system failure. One is based on system data alone, such as in n reactor-years of experience there have been f system failures, or in d system demands there have been f failures. This is the approach of WASH-1270. The other approach is to separate the system probability into its components, such as hardware, test and maintenance, and human error; estimate the probabilities associated with each component; and then add. This is the approach of WASH-1400 and followed by GE in their ATWS analyses. EPRI takes both approaches, then merges the results.

2. Some other differences: The assumption in WASH-1270 is that scram system unreliability, call it Q_{WS}, is constant across time and across reactors. In WASH-1400 it is assumed, I believe, that Q_{WS} varies among reactors, but not over time, according to a specified distribution. EPRI goes one step further and assumes that rod failure probability varies from rod to rod in one reactor at one time. GE, I believe, regards Q_{WS} as a constant, but uses WASH-1400 methods anyhow. All this makes comparison of results difficult. For example, the upper 95% confidence limit on Q_{WS}, by WASH-1270 methods, is a bound on the unknown, but industry-wide, value of Q_{WS}. If it is assumed that Q_{WS} varies among reactors, but not very much, then the WASH-1270 95% confidence limit can be regarded as an approximate 95% confidence limit on the average Q_{WS}. The upper limits of WASH-1400, in contrast, are to be interpreted as bounds for 95% of the reactor population. That is, in one case we have a statistical bound on a population average; in the other, a probabilistic bound on population individuals. The numerical value of these bounds should not be expected to be comparable nor should coincidence of the bounds be taken as confirmation or support of one analysis for the other.

3. While the system synthesis approach of WASH-1400, EPRI, and GE have the potential for providing more precise estimates than the WASH-1270 approach, that potential is not realized because of flaws in the methodology which make the results highly questionable. Attachments A and B describe in detail the quantitative errors which result from the "square root" method of probability estimation used in these three analyses.

4. The errors the square root approach leads to are that failure probabilities are understated, relative to the assumptions on which they are based. It may be that the assumptions in these analyses are quite conservative and that the analysis method just offsets this conservatism, but this cannot be relied on. Before we can put any trust in numerical results, we must be able to trust the methods used to obtain those results. Incorrect probability equations are as much of a problem or danger as incorrect heat transfer or fluid dynamic equations.

5. The issue is not one of approach - Bayesian vs non-Bayesian vs empirical Bayesian, or statistical vs risk analytic, etc. - but one of mathematical correctness. Given a problem, given some assumptions, how is the answer to be derived? The mathematical rules of probability provide the answer. Ad hoc approaches which violate those rules do not.

C. <u>Probability Estimates</u>

1. <u>From System Data</u>
Under the not necessarily conservative assumptions that the system failure rate has been constant across the accumulated 2220-reactor years and continues to be, and that 1 failure has occurred, the maximum likelihood estimate of the system failure rate, θ_{WS}, is $\hat{\theta}_{WS} = 1/2220 = .00045$. Upper confidence limits on θ at the 75, 95, and 99% levels, respectively, are .001, .002, and .003. Corresponding lower confidence limits are 1×10^{-4}, 2×10^{-5}, and 5×10^{-6}. Under the nonconservative assumption that the time between transients is exponentially distributed with rate μ and the assumption of monthly testing which would detect and correct failures, ATWS probability is approximately

(for small $\mu\theta$) equal to $\mu\theta/24$. The following table gives upper and lower confidence limits on ATWS probability, given $\mu = 10$.

ATWS Probability

Confidence Level	Lower Limit	Upper Limit
75%	4×10^{-5}	4×10^{-4}
95%	8×10^{-6}	8×10^{-4}
99%	2×10^{-6}	1×10^{-3}

If only central power stations are considered, all these estimates would double. If no failures are assumed, the upper 75, 95, and 99% confidence limits on ATWS probability become, 3×10^{-4}, 6×10^{-4}, and 8×10^{-4}, respectively, little different from the upper limits assuming 1 failure, and the lower limits all equal zero.

An alternative to assuming constant system failure rate is to assume a constant probability of failure on demand. Under the nonconservative assumption that there have been 114,332 independent demands, all of which had a probability of failure, P_{WS}, and only one failure, the maximum likelihood estimate of P_{WS} is $\hat{P}_{WS} = 1/114,332 \approx 10^{-5}$. Under the same assumption as above concerning the arrival rate of transients, ATWS probability equals $1 - \exp(-10\ P_{WS})$. From confidence limits on P_{WS}, the following limits are obtained on ATWS probability:

ATWS Probability

Confidence Level	Lower Limit	Upper Limit
75%	3×10^{-5}	2×10^{-4}
95%	5×10^{-6}	4×10^{-4}
99%	9×10^{-7}	6×10^{-4}

If no failures are assumed, the upper limits become 1×10^{-4}, 3×10^{-4}, and 4×10^{-4}. Note that the estimates obtained in the constant failure probability case are about 1/2 times those in the constant failure rate case and the assumption of monthly testing. Test frequency plays no direct role in the constant failure probability case.

2. From Component Data
Attachment B provides estimates of multiple control rod failures, conditional on successful operation of the reactor protection logic. The estimates are based on simple models, chosen to fit the EPRI data. The models are not particularly conservative because they entail rather smooth, constrained variability.

To estimate the conditional system failure probability from the results in Attachment B, a definition of system failure is needed. For BWR's, EPRI and WASH-1400 define it as failure of three adjacent rods. GE says 5 adjacent rods. To estimate the probability of these events, the probability of adjacency, given that x rods fail, is needed. One approach is to assume all sets of x are equally likely and then count how many of those yield 3 or 5 adjacent rods. EPRI and GE take this approach. As a simpler, but no more arbitrary (in the absence of any data pertaining to the probability of adjacency) approach one might consider a function of the form,

$$\text{Prob(a adjacent failures} \mid \text{x failures)} = 1-\delta^{b(x-a)+1}, \quad x=a, a+1, \ldots$$
$$= 0 \qquad , \quad x=0, 1, \ldots, a-1.$$

For the models in Attachment B, this leads to

$$\text{Prob(a adjacent failures} \mid \text{scram attempt)} = \text{Prob}(x \geq a)\left[1-\frac{(1-\rho)}{1-\rho\delta^{b}}\right],$$

where ρ is the bracketed term on page 12 of Attachment B, namely $\rho = p\theta/[1-(1-p)\theta]$.

Consider the case of a = 5. GE estimates that the conditional probability of adjacency, given 5 failures in a 177 rod core as 10^{-5}. Thus, to coincide with this, one obtains $\delta = 1 - 10^{-5}$. To choose b, we use the GE results that system failure occurs conservatively with probability of .05 when 55 rods fail. Equating .05 to the probability of 5 adjacent failures, given 55 failures, leads to b = 100. Using the maximum likelihood estimates of r, θ, and p in Attachment B thus leads to

$$\widehat{\text{Prob}}(\text{system failure} \mid \text{scram attempt}) = 2 \times 10^{-7},$$

Using the upper 95% confidence limits on r, θ, and p leads to

$$\text{Prob}_{95}(\text{system failure} \mid \text{scram attempt}) = 8 \times 10^{-6}.$$

These results are highly conjectural and would require considerably more study. For example, the fact that there may be some previously failed rods present has not been accounted for. However, these results are no more conjectural than previous analyses and are at least consistent with available data and with the rules of probability.

For PWR's, failure is defined by EPRI as failure of 3 or more rods. An upper, approximate 99% confidence limit on the probability of this event is, from Table 3 of Attachment B, 1.5×10^{-3}. Thus, the PWR data, which are much more limited than the BWR data, and the less stringent system failure definition, lead to a considerably higher bound than that of BWR's.

D. Conclusion

There is too little information available right now to make a reasonably precise assessment of system failure probability from available component data. This leaves only the systems data on which to base a quantitative assessment. Available data do not resolve the question of whether an ATWS fix should be electrical, mechanical, or both.

Attachments: (Available on request)

Attachment A - Failure Probability Calculations for GE ATWS Analysis

Attachment B - Estimation of Control Rod Failure Probabilities

IV. Equipment Reliability

Low probability single failures not required; however, if the mitigating system unreliability was significantly greater than 10^{-3} per demand, analyses assuming these failures were required. The data source essentially was WASH-1400 median values. For estimating population risks, selection of median values as compared to mean values may be nonconservative.

V. Assumptions and Input to Evaluation Model

Proposed ANS Industry Standard N661 on Anticipated Transients Without Trip on PWR Plants recommends the following on plant conditions and assumptions for evaluation of ATWS events.

The value used for each condition and assumption shall be selected by one of the following methods.

1. Selection of a conservative value as specified or defined by either the design basis FSAR analysis, or the technical specification limit.

2. Selection of the design operational value allowance for control band, but excluding any allowance for measurement uncertainty, for variables regulated either by automatic control systems or manually under administrative control.

3. Selection of either the measured or design value excluding any allowance for design margin or measurement uncertainty.

4. Selection of a calculated value not expected to be exceeded (that is, more adverse) during the preponderance (at least 95%) of plant lifetime. Justification of this probability argument need not consider allowance for calculational uncertainty or for random statistical fluctuations.

In general, vendors used values consistent with 2 and 3 above except for MTC where 4 above was used as basis.

The present staff ATWS model input requirements are consistent with the standard except in the following areas.

a) Instead of 95% moderator temperature coefficient (MTC) use 99% value. Transient frequency is high when MTC is high. See note from A.Thadani to S.Hanauer dated 3/8/77 for more details on this requirement.

b) Purging in Progress during ATWS event. This requirement was imposed because of the staff belief that while some plants have frequent purging, others may undergo continuous purging. In any case, this requirement does not appear to require any design modification.

c) Ten percent primary safety valve accumulation to open for water discharge. Although the standard implies use of three percent accumulation, discussions with valve manufacturers indicate the ten percent value to be more realistic. Vendors seem to agree with us after discussing the problem with valve vendors.

On the other hand, the staff has not required inclusion of:

i) over ½ GPM Steam Generator leakage

ii) coincident loss of offsite power

iii) uncertainty in operating parameters

iv) any conservatism beyond 0.9* Homogeneous Equilibrium Model for primary system water relief through relief and safety valves (data lacking but the staff judges the model to provide low estimates on water relief)

v) no operator error

vi) no operator action in the first ten minutes of an ATWS event

vii) Seismic Events

VI. Effect of ATWS fix on non-ATWS Accidents

BWR:

WASH-1400 Significant Core Melt Sequences

Category 3

TW-γ	1×10^{-5}
TC-γ	1×10^{-5}

Category 2

TW-γ'	3×10^{-6}

Category 1

TW-α	2×10^{-7}
TC-α	1×10^{-7}

$T \equiv$ Transient Event
$C \equiv$ Failure to Shutdown Reactor
$W \equiv$ Failure to Remove Residual Core Heat
$\gamma \equiv$ Containment Failure due to Overpressure-Release through Reactor Building
$\gamma' \equiv$ Containment Failure due to Overpressure-Release direct to atmosphere
$\alpha \equiv$ Containment Failure due to Steam Explosion in the Vessel

WASH-1400 suggests that two types of accidents i.e. TC (ATWS) and TW (failure to remove decay heat) are major contributors to core melt.

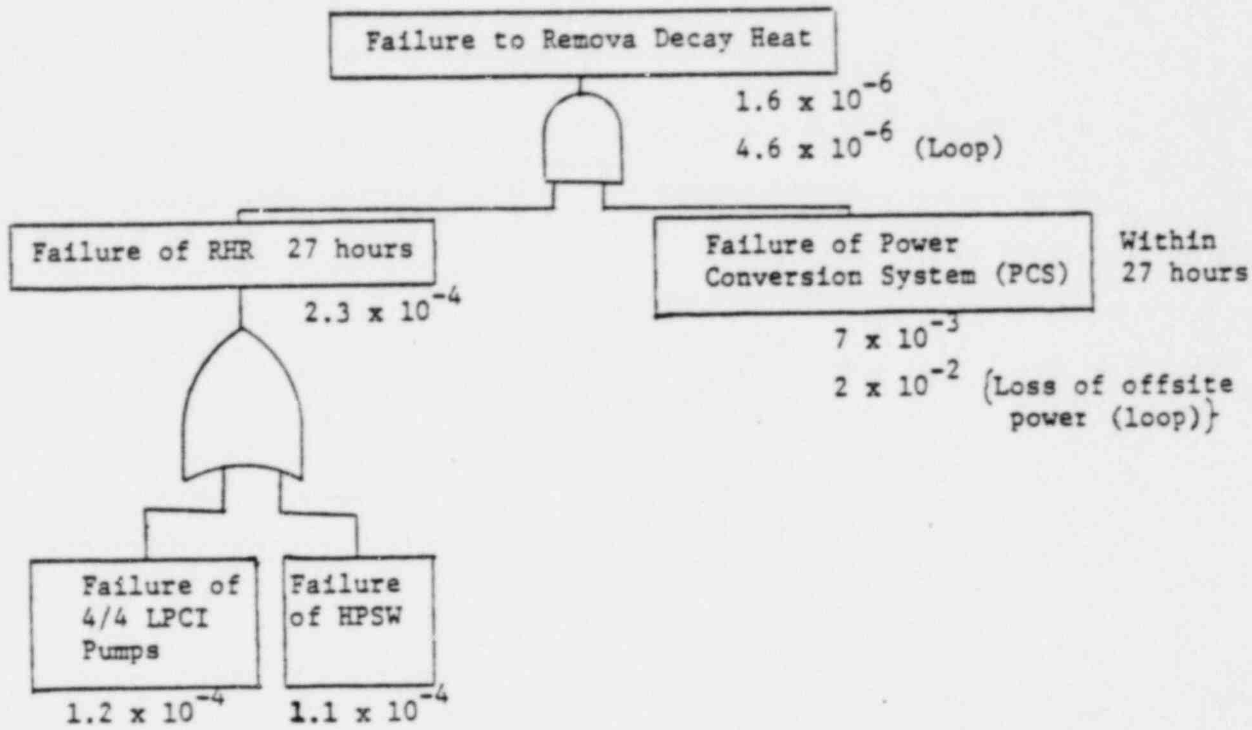
As discussed elsewhere (note from A. Thadani to S. Hanauer dated April 8, 1977).

The probability of unacceptable consequences due to ATWS is 10^{-4} per reactor year and one of the indicated design 'fix' was to add a high pressure special ATWS make up system (SAMS).

The failure probability of the decay heat removal system (W) is dominated by the failure probability of power conversion system to perform the function of transferring fission product decay heat to the environment. A careful consideration of this failure mode in the design of SAMS is expected to result in lower probability of core melt from TW sequences.

BWR

RESIDUAL HEAT REMOVAL SYSTEM



Note the probability of Core Melt due to Non-Loss of Offsite event $\sim 10 \times 1.6 \times 10^{-6} \sim 10^{-5}$ dominating over loop event. Success of Power Conversion System (PCS) depends on ability to

a) Operate one complete condensate feedwater piping i.e. condensate and feedwater pump.

b) Open one Isolation Valve and open bypass valve.

c) One condenser recirculating pump operable.

It is not clear in WASH-1400 what the individual contributions to PCS unavailability are but the contribution of a above would be expected to be reduced by availability of SAMS.

PWR:

WASH-1400 Significant Core Melt Sequences are:

1. Small LOCA followed by failure of containment and core cooling systems.

2. ATWS

3. Check Valve - Interfacing System LOCA

4. Transient followed by failure of main feed and auxiliary feedwater system.

Comments:

The Interfacing System LOCA probability has been reduced by recent NRC requirements.

It is not clear why transient followed by failure of main feed and auxiliary feed is assumed to result in core melt. In any case ATWS fix may help reduce this probability by cooling the reactor using high pressure injection system and pressurizer relief valves.

ATWS probability resulting in exceeding criteria is believed to be higher than the WASH-1400 estimate. Thus, if ATWS fix is provided the core melt probability may be controlled by small LOCA ($10^{-5} \sim 10^{-6}$).

Preliminary Considerations -

ATWS vs. Use of Faulted Stress Limit

The Faulted Limit permits primary membrane (average stress across vessel wall) stress levels considerably in excess of the yield strength of the material.

A. Reactor Vessel

Discontinuity regions of the vessel could be expected to deform plastically, i.e. they would not return to their original shape after load relaxation. Such regions would include intersections of primary coolant nozzles to vessel shell, flanges to shell, and head dome to vessel shell or to closure head flange. Regarding the latter item, the extent of the resulting permanent deformation could conceivably be great enough to prevent manual insertion of at least some of the control rods. The effect of vessel distortion on the position of the control rod blade passages relative to their design location would also have to be considered.

Another major consideration to be evaluated would be the behavior of the bolted closure head to vessel flange joint. At such high pressures leakage would probably be severe, i.e. the bolts are only torqued for normal operating pressure. What would be the effect of severe coolant leakage on such things as local fuel rod overheating etc.? Regarding the behavior of the vessel to head juncture and possible head distortion MEB currently has a contract with INEL to evaluate the effects of a pressure of 3750 psi on these areas of a B & W 171 FA reactor vessel.

In light of the ongoing ATWS re-evaluation, perhaps we should have INEL evaluate the effects of 4500-5000 psi instead. Structural integrity of instrumentation tube and control rod drive housing partial penetration welds would have to be evaluated. Section III of the ASME code only allows the use of such welds when the penetration nozzle is subjected to essentially zero piping reactions. Pressures which result in faulted limit stresses and consequent deformations of both vessel heads some of these welds may fail - consequences must be evaluated. Instrumentation probes and any control rods in core could be ejected.

B. Pumps and Valves

Active Valves

Valves of this type which have to function after the ATWS so that ECCS, CVCS etc. systems can be brought into play to shut down the plant would experience large permanent deformations. Their capability to function after exposure to such stress and strain levels could probably only be convincingly verified by test. Additionally, the comments below for inactive pumps and valves would apply.

Inactive Pumps and Valves

From a structural integrity point of view, these components, for the most part, can be likened to small stainless steel pressure vessels.

Permanent deformations in regions of discontinuity would be severe. The extent would have to be evaluated on a component by component basis.

Behavior of large bolted bonnets of these components, again, as for reactor vessel only torqued for 2500 psi, is open to question and would have to be evaluated.

C. Pressurizers

Permanent deformations and their effects would have to be evaluated.

Areas of concern - bolted manway covers and pressurizer heater to bottom head welds (burst pressure of these welds could be exceeded).

D. Steam Generators

Effects of pressure per se on tubes would probably not be a problem. However, again, parts of this vessel would deform severely. The effects of such deformations on "tender" spots such as tube to tube sheet weld integrity would be difficult to evaluate with any degree of confidence.

Also bolted manway covers could be a problem.

E. Safety and Relief Valves

Valves could be expected to open and relieve because this occurs, for ATWS transients, at normal valve set pressure. Question here would be to what degree would they reclose after experiencing some deformation. From discussions with safety valve manufacturers, it would appear that from basic structural integrity point of view, the valves should be o.k.

F. Piping

Exposure to pressures which result in the stresses at the faulted limit would not be expected to result in any breach of piping from this effect alone. However, plastic deformation of other components in the system, together with that in the piping would result in increase of nonpressure mechanical loads in various locations in the piping, such as at intersections with other large components, elbows etc.

The effects of the high pressure in combination with the other resulting mechanical loads on the dimensional stability of the piping would have to be evaluated. For instance, even though the pressure boundary of the piping may not be violated, dimensional distortion of piping at a critical location could impair flow of fluid from ECCS, CVCS etc. alternate shutdown systems.

General Remarks

The use of the Faulted limit allows for considerable inelastic deformation of the material. Permitting the use of this limit for pressure loading means that for the first time NRC would permit the entire reactor coolant pressure boundary to deform plastically for a postulated event. Thus far this stress limit has only been applied for relatively concentrated high load situations i.e. LOCA, LOCA plus SSE and always for events where the control rods are in. Although we say that we will accept the Faulted Limit for SSE alone; in general, the majority of components under just the SSE environment are not exposed to stress levels anywhere near the Faulted limit

In order to fully evaluate the consequences of going to the Faulted limit on pressure loading, all components would have to be analyzed inelastically. Then a complete system inelastic analysis would have to be performed to determine the effects of the component deformations on the piping. NRC would have to review all such analyses. From the vendor's point of view, going this route would be expensive and would not necessarily result in an NRC approval that such high pressures would have no adverse impact on the public.

Also in performing these types of analyses, it is not always possible to accurately define the actual magnitude of the loads to be input to the analyses. Thus it is not clear at this time that with all of the unknowns and uncertainties involved in going to such a high stress limit, on such a gross basis, that safe shutdown of the plant could be demonstrated within the level of confidence NRC would require.