UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555

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MEMORANDUM FOR: Roger Mattson, Director, Division of Systems Safety

FROM: Victor Stello, Jr., Director, Division of Operating Reactors

SUBJECT: COMMENTS ON THE STAFF'S DRAFT ATWS REPORT

The Division of Operating Reactors has reviewed the Draft ATWS Report and is providing comments in the following manner: Attachment A contains detailed comments developed by the DOR technical staff (a copy was supplied to A. Thadani on November 7). Attachments B, C, D and E, which were developed at the Branch Chief level in DOR, contain comments related to policy considerations associated with ATWS.

Our principal comments on the report are highlighted below:

(1) The recommendations in the report are not clear. Specific policies and deterministic acceptance criteria are required for each class of plants.

(2) Timely and effective implementation of ATWS requirements will hinge on acceptable analytical models. Balanced decisions as to the requirements themselves are similarly dependent. It does not appear that model development has progressed sufficiently to satisfy these considerations.

(3) The selection of transient probabilities and failure to scram probabilities (e.g., for BWRs) may result in requirements which place a large burden on these plants without the support of a rigorous or compelling probabilistic basis. Further efforts may be needed related to the confidence levels (e.g., 95-95) associated with each of the event probabilities to ensure their appropriateness and consistency.

(4) While not purported to answer the question of backfitting, the report does not provide a suitable framework for making future backfitting decisions.

We recognize that the nature of these comments are quite significant and that it will be a lengthy and tedious task to revise the report, we have

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made arrangements for Mr. R. Cudlin, who orchestrated our comments, to be available to work with your staff toward the goal of finalizing the ATWS report.

Victor Stello, Jr., Director
Division of Operating Reactors

Enclosures:
As stated

cc: S. Hanauer
A. Thadani
D. G. Eisenhut
K. R. Goller
DOR Branch Chiefs
R. Cudlin
J. Guibert

Attachment A

DIVISION OF OPERATING REACTORS

DETAILED COMMENTS ON "TECHNICAL REPORT ON

ANTICIPATED TRANSIENTS WITHOUT SCRAM"

(Draft of October 17, 1977)

P. 2-23 References are made to 10 CFR 100 dose guidelines and Iodine-131 releases. It should be specified that these are thyroid doses from inhalation of radioiodine. No estimates appear to have been made of whole body doses due to immersion in cloud.

2-23, VI changed

P. 5-8 It is stated that the failure of steam generator tubes "is not serious unless there is considerable fuel failure in the transient." The word "considerable" should be deleted.

Page 5-8 changed

Parts B, C, D, E, Section 2.3.5

The emphasis on assumptions that lead towards maximizing the primary coolant pressure may not be the most conservative from the point of view of radiological consequences as long as vessel integrity is maintained. For example, the assumption of pressurizer relief valves remaining closed will increase the system pressure but decreases the releases of primary coolant to the containment and, thereby, the offsite doses. Delaying isolation of a faulty steam generator or prolonging primary to secondary leaks may also need to be considered.

P. B-64 It is stated that "the secondary side offsite dose is approximately linear with primary to secondary leakage rate." This is incorrect. As shown in Figure 1 of Appendix VI, the doses are linear only if plotted on log-log paper.

Changed page B-64

Appendix VI

Although no quantitative estimates of the effect of fuel failures are given, the general tone of Appendix VI seems to indicate that the radiological consequences would not exceed the 10 CFR Part 100 guidelines even with failed fuel. This is misleading because the presence of failed fuel may be quite significant. For example, for a steam generator leak of 100 gpm, over 10,000 Curies of I-131 would be released over 30 minutes with only 5% of the fuel failing. This is in excess of the 10 CFR 100 guidelines even with very good meteorological dispersion. With 10% failed fuel, the guidelines could be exceeded with leak rates as low as 10 gpm for 30 minutes.

In addition, the report neglects to consider the cases where the primary coolant iodine activity exceeds the equilibrium activity limit because of a spike produced by an earlier transient. The current Standard Technical Specifications for PWRs allow the primary coolant iodine activity to be as high as 60 Ci/g at 100% power for 10% of the operating time. An ATWS under these circumstances could exceed the 10 CFR 100 guidelines (a 100 gpm leak for 30 minutes resulting in a release of over 500 Ci).

It is stated on Page VI-8 that if the steam generator is flooded, "the estimated dose can be determined by reducing by a factor of 10 the value obtained from the curve" in Figure 1. This is incorrect and not conservative because the releases under these conditions may be several times higher.

Section 3.3.2.5

It is recommended that site dependent dispersion factors be used in the analyses of the radiological consequences, but the probability level to be used is not specified. In order to effectively implement this recommendation, it would be necessary to set a specific probability of occurrence for the meteorological conditions to be used. It should be noted that this can be a determining factor in meeting the acceptance criteria. For example, the 5% conditions typically result in consequences a factor of 10 higher than those resulting under 50% conditions.

Appendix V and Section 3.3.2.4

In addition to the determination of a leak rate, there is a need to estimate the duration of the leak, and the means to stop it or mitigate its effects, e.g., reflooding the steam generator. This area is not ddressed in the report although a duration of 30 minutes is used for the calculations in Appendix VI. This is a critical area (see reference 7 to Appendix V) that should be taken into account when calculating radiological consequences.

For Category 2 plants (new plants) it is recommended in Appendix V that a "steam generator primary/secondary leakage rate equal to the techical specification upper limit for restricted operation by assumed for ATWS evaluation." Section 3.3.2.4 recommends twice this leak rate. No justifiction is given for either of these criteria.

P 1-10 It is questionable whether or not the reliability goal of 10 - 10 can be achieved by single non-IEEE-279 ATWS mitigating systems. The NRC review of the reliability goal for these ATWS mitigating systems will require a probabilistic assessment in order to verify that the reliability goal has been achieved.

Done

P 1-11 Table 1.2 - Staff ATWS Requirements. This staff requires that the diversity of mitigating systems must be demonstrated. Since there are no NRC criteria for acceptable means of achieving diversity, a section needs to be added which provides typical means for achieving diversity which are considered acceptable by the staff.

Done

P 1-11 Table 1.2 - Staff ATWS Requirements. B&W and CE applicants are required to provide all operator actions and the time into the ATWS event when these actions are to be performed. Table 1.2 does not include these requirements for GE and W applicants. It is recommended that Table 1.2 be modified to include these requirements for GE and W.

P. 3-10 The argument that the staff will use RSS data and evaluations to establish the probabilities for each single failure mode unless a most compelling case is made for other values is questionable. Although the RSS data is assumed to be better than most, it should not be blindly used without consideration of more current data and evaluations that may be proposed.

Done W.M.

P 3-12 We believe that 20 minutes should be used for the minimum time at which reliance on operator action is allowed.

page 4-6 Covered A15 SN

3-37 In view of the recent steam generator tube leakage experience and the present uncertainty in determining the cause and the cure of all steam generator tube leaks, it does not make sense to recommend two different methods of including steam generator leaks in the ATWS evaluation i.e., one for new plants and plants with little or 'small amount of routine leakage' and one for operating plants for which there is a continuing significant steam generator tube leakage problem.

Changed page 3-38 V-S

P 3-46 Add the following additional means of achieving diversity:

e. Use of different logic equation.

f. Use of open or closed logic contacts during operation.

g. Use of separate location of logic i.e., RPS cabinets and ECCS cabinets.

AT?

D-40 The stability of the reactor core in response to an event wherein the GE recirculation pumps are tripped is of concern to the staff because of (1) the occurrence of high clad surface heat flux to flow ratios encountered in the calculated response to a RPT, and (2) the reduced stability margins at high power to flow ratios at natural circulation. Since NRC is currently requiring RPT on all operating BWRs it is recommended that the above staff concern be resolved ASAP. NRC would be in an unenviable position if after requiring installation of RPT on all operating reactors, it was decided that RPT is no good.

E-69 Westinghouse's analysis assumes that there is automatic initiation of the turbine trip, automatic initiation of the auxiliary feedwater system, and automatic initiation of containment isolation. However, the staff notes that for Westinghouse plants the systems which initiate turbine trip, auxiliary feedwater, and containment isolation are not necessarily diverse from the reactor protection system. Consequently, a common mode failure of the reactor protective system could lead to loss of these three functional systems. Without this diversity and automatic turbine trip, the primary system peak pressures will exceed the design limits and late initiation of auxiliary feedwater may lead to core melt. It is therefore recommended that staff requirements be added to require the systems which initiate turbine trip, auxiliary feedwater and containment isolation to be diverse from the reactor protection system to reduced the probability of a common mode failure to an acceptable level.

III-1 Tables 1 through 4 present rod drive failure data. It is recommended that a discussion/analysis of the data be included in Appendix III. Without a discussion, the reader is apt to draw erroneous conclusions from the 27 pages of rod drive failure data.

VII-10 The selected probability distributions for each of the reactor operating parameters i.e., 'uniform' or 'normal' distributions were used for convenience because the paucity of data and potential biases did not seem to justify more complication functions. In lieu of cautioning the reader in using the results of this study because of sizeable inaccuracies, uncertainties and parameters not included in the study, one has to question the worth of Appendix VII in the ATWS report. Perhaps it should be removed.

Section 2.3.2 of Parts B, C and E

The report indicates that the reactor coolant pressure boundary can be exposed to a pressure up to 3200 psig (presumably 3200 psid, pressure difference rather than gauge pressure) without exceeding the "Emergency" condition stress limits. These calculations were not correct in view of the fact that most PWR steam generator tube plugging criteria are determined such that degraded tubes are allowed to be stressed up to the yield strength during the full range of normal operating condition.

During ATWS conditions, the maximum differential pressures for B&W and CE designs exceed 3750 and 3200 psid, respectively. The "Emergency" condition stress limits will probably not be met even without considering tube wall thinning. For Westinghouse design, the requirements for meeting the ATWS criteria will result in such a small maximum allowable tube wall degradation, it may not be possible to detect with the present state-of-the-art eddy current testing (ECT) method.

With the general trend in the industry to go with thinner tube wall design, the requirements to meet ATWS criteria will have substantial impact should some forms of degradation appear.

For most operting PWRs that have experienced tube wall degradation, the requirements to meet the proposed ATWS criteria will drastically reduce tube plugging limits. For some units, this will result in severe economic penalties.

In most ATWS evaluation models for PWRs, the heat removal rates of steam generators appear to exclude the effect of loss of heat transfer surfaces of plugged steam generator tubes. For instance, the CE model using Calvert Cliff units may be representative among CE operating units because Calvert Cliff units are relatively new and have not had tubes plugged (or very few if any at all).

On Pages 2-110 through 2-116, it was indicated that, among three alternatives for improving the reactor system to meet ATWS requirements, the third one to improve plant response to ATWS was the staff's preferred approach. On the basis of the above comments related to steam generators and assuming some continued forms of steam generator tube degradation, that conclusion could change.

Comments related to the "Emergency" condition stress limits for ATWS condition.

The ASME Code does not provide explicit rules to address such phenomena as corrosion and stress corrosion that may result in cracking and other forms of material degradation and that Appendix G does not have explicit rules to accommodate accident or transient loadings. The pressure loading associated with ATWS has a duration such that every component within the reactor coolant pressure boundary experiences this pressure as essentially a static pressure. Therefore, the stress induced during the ATWS condition that meet the "emergency" condition limits may result in permanent deformation in the reactor coolant pressure boundary with Ausenitic materials.

B-4 The staff normally specifies a discharge coefficient for safety valves of 0.97.

B-18 Reference is made to the staff sensitivity studies of varying auxiliary feedwater initiation time. However, no results of the study are presented.

Changed B-18

B-36 What is the significance of conclusion 1 given the number of similarities between the two computer models (e.g., moderator reactivity feedback, steam generator heat transfer correlations, etc.)

P. B-57 (Changed B-57) The third column in Table B-2 should be peak clad temperature.

P. B-60 (Changed B-60/ch3) Why is B&W being required to look at the affects of creep collipase, and not the other vendors.

P. C-7 What is meant by "practicably the same."

P. C-16 (Revised text) It is not clear whether or not CE used the homogenius equilibrium model.

P. C-21 (Revised text) It is reported that the steam generator safety valves lift at 895 psia. Page C-17, however, specifies that these valves will open at 1000 psia.

P. C-27 It is stated that staff has reasonable assurance that the results of upcoming startup tests will not materially affect the CE ATWS model. This statement in itself is unsupported and should probably be restricted to the effect that the staff does not anticipate that he results upcoming test will materially affect the codes.

P. D-6 Bottom on Page. It may not be conservative for ATWS to overestimate steam flow.

P. D-31 (Will Change) At the bottom of the page it is stated that the point of operation with the least margin of stability is at the 105% rod line and natural circulation flow, i.e., point A of Figure D.1. How can this be a point of operation since it is above the APRM rod block line.

P. D-35 What is meant by the third order feedback system.

P. D-36 What is meant by a limit cycle.

P. D-40 What is meant by integrated energy. This term has not been previously discussed.

P. D-40 Conclusion 2-A is not consistent with Item 3 on Page D-35.

P. D-41 Conclusion 2-C is not consistent with Conclusion 2-A

P. D-57 Why is failure mode (failure of safety valves to open) assumed for PWRs but not for BWRs.

P. D-57 The last paragraph indicates that recirculation pump trip may be something of a useless gesture in satisfying the staff ATWS safety goal. This is not consistent with current staff requirements on BWRs.

P. E-21 The moderator density coefficient used by Westinghouse is more negative 97% of the time whereas the MDC used by CE and B&W is more negative 99% of the time.

P. E-60 The last paragraph should specify which plants' evaluation model is applicable.

P. II-8 The last paragraph criticizes the assumption of independence of rod scrams yet our model makes that very same **assumption** (See Page II-4).

P. II-19 Is the criticism of point estimates to be construed as a personal opinion, or as NRC policy.

P. V-4 It appears that new PWR plants should be in Category 2 rather than in Category 1.

P. IX-2 A discussion of the required 0.97 multiplier should be included here.

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A.T.

November 22, 1977

(24)

Note to A. Thadani

SUBJECT: ATWS

Here are my initial comments on the topics requested in our meeting last Friday.

1. Boyd, p. 2, top paragraph. This is correct. It appears from the analysis in Appendix VII that the Status Reports may not have been severe enough to achieve the 10^{-7} goal. I am frankly a little hesitant about evaluating whether this is really true, or whether, as I suspect, the analysis in Appendix VII simply doesn't get into this area sufficiently. I tend not to believe the curves and numbers in Appendix VII beyond the order of 10^{-6}, which is about 1% conditional probability given a severe ATWS. I would have to at least think some more and maybe do some more calculations before working in the 10^{-7} area. Furthermore, it's not clear to me that the response surface is a good approximation to peak pressures in that range. I would be surprised if the linear approximation still holds. For example, I would prefer to say that we simply haven't done enough calculations to know what it takes to achieve 10^{-7}.

already answered GK

The goal of Standard Review Plan 2.2.3, while not discussed in the referenced comment, is perhaps worthy of discussion and inclusion in any discussion of probabilistic ATWS goals. They are not the same as the WASH 1270 goals, since they are expressed as 10^{-7} realistic or 10^{-6} conservative. Neither is this goal the same as the one now proposed in the Draft ATWS Report. I agree with Boyd that more discussion in this whole area would be appropriate.

2. Boyd, page 2, paragraph 2. This comment is basically correct. The probabilistic analysis and the probabilistic goals and the probabilistic discussion, and all of the discussion of WASH 1400, both in establishing goals and in discussing events sequences, are by way of attempting to establish a deterministic licensing procedure on a more rational basis, but indeed, in the present state of the probabilistic art, we have to rely finally on judgment. The objective is to attempt to quantify some of this judgment and to try to make our thinking processes as clear and rational as we can in the face of uncertainties. We don't know what is going to happen and the values of parameters have a range, and also there are uncertainties inherent

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in our present difficulties in thinking our way through from probabilistic to deterministic goals. If you look hard enough, you will discover that our present deterministic goals are also based on some notions of probability, as I indicated starting on the bottom of page 2-5 in the ATWS draft.

3. Denton, unsigned draft, Comment No. 1. This is a complaint rather than a comment. Sure we have problems in this area. That's why we try to write an ATWS report. If we wait until everything is perfect and approved, we will never make any progress because so many different papers in the area must necessarily be related. An alternative approach is to make what progress one can in the topic at hand with the realization that perfection will not be achieved. A more productive point of view along this same line would be to recognize, as in Item 2 just above, the difficulty in the present state of technology of establishing probabilistic or deterministic goals.

No

The use of WASH 1400 in this connection is, I think, both productive and acceptable, but I would not for a moment suggest that it is without problems. The objective is to define what it is we are doing with WASH 1400 and other things, and to make sure that we don't try to do too much with it.

SN

4. Denton, unsigned draft, No. 2. Well, there is some conflict and this is recognized, I think, in the ATWS draft. Maybe it ought to be recognized more explicitly. To state that it is an issue doesn't necessarily help in resolving the issue. I guess I would like to know what Bunch, Denton, and company think ought to be the resolution.

5. Denton, unsigned draft, No. 3. Again, yes, of course. The key word is reliability or event sequence probability. The proposed method of analysis flies directly in the face of the "safety equipment only" and "single failure criterion" approaches so often used. I think this is a useful and potentially productive departure from today's practice which has of course to be considered very carefully and validated as we go along. The present rules have led to many absurdities, and it is time to see if we could do better.

6. Denton, unsigned draft, Item 4. Again, yes, indeed; I see a big issue. The whole idea here is realistic evaluation including obviously a realistic assessment of the error bands and uncertainties. Let's at least give it a good try before we fall back on the old standby.

O.K.

7. Grimes, A1. Nonsense. We have said in a whole lot of ways, and on a whole lot of occasions, that a threat of this magnitude requires fixing in good order and good time, but not an urgent requirement. If by "BWR implementation measures" Grimes means the recirculation pump trip, then he is right, and we ought to get off that particular dime. If he means not taking nine more years, he is also right. If he means shutting them down, or having some crash program, I think he is wrong.

8. Grimes, A2. I think this comment is probably correct. The detailed technical parts of it should be put into the writeup as appropriate and Grimes' help should be solicited in rewriting both the report and the summary to reflect what is significant about this comment.

9. Grimes, A3. This comment may be correct technically, but it will take us years to know whether it is true. The Reactor Safety Study is the best we've got. I have taken it here as exemplary of the realistic approach to reactor safety and risk. Only in the area directly involved with ATWS have I looked behind the study, and made what corrections I thought were appropriate. I'm sure corrections could be made to improve the safety study in other areas and for other sequences. However, it's too much work, and I would guess that the result would not, overall, be very different. I don't think that we could do it any better than the Safety Study staff did, and likely not as well.

10. Grimes, A4. I don't agree with this at all. The issue here is the concept of "acceptable risk." I've soft pedalled that in the ATWS report, because it is so difficult and controversial, and didn't seem to be needed. However, this comment of Grimes highlights the problem, and maybe the problem should get more attention. If reactor safety is at an acceptable level, then it still may be cost effective to make some inexpensive change which improves safety by a moderate amount (since there is only a moderate amount of improvement left.) However, I don't think it is worth 60 pages, much less 600 to do any detailed exploration in this area once an acceptable risk has been reached. The point of the ATWS report is that the present risk is not acceptable in the long range for ATWS for BWR's and for PWR's not like the ones in the safety study, and therefore, a fix is required. While the door is always open to future improvements and future insights, I strongly reject this notion of Grimes that every time we think of something new that is cost effective that we are going to impose it on everybody. However, I do tend to agree with Grimes that ATWS arguments based on the risk from other contributors is less strong

than the one based on acceptable risk. If we could find something good to write about on acceptable risk, I would be in favor of slanting the section more in that direction.

11. Grimes, A5. It's not clear to me whether Grimes is using highly conservative design basis meteorology or the realistic meteorological spectrum which one ought to use if one is doing realistic calculations. This comment is in an area I have not done any direct study on, and should be worked on by people who know more about accident calculations and their results than I do. We should obviously say what is technically correct.

SN I agree

12. Grimes, B2. I think this is nonsense. No showing of grid reliability over 40 years would convince me of anything except the incompetence or untruthfulness of the analyzer. If Grimes doesn't like the present review of approach in licensing, he ought to fix it and not make an ATWS Report his whipping boy.

SN

13. Grimes, B5. That's a good idea.

working on it.

14. Minners, 2-42. The test frequency is discussed at some greater length on page 2-73. I think the whole subject does indeed deserve a more serious, more concentrated, and more comprehensive treatment.

– Incorporate in Ch 2 rewrite WM

15. Minners, 2-104. This comment is not correct. We don't (or shouldn't) criticize GE and EPRI for using the log normal distribution, but for using the square root for approximating the common mode failure contribution. Log normal seems to be the best thing available to use to approximate many frequency distributions of use in this area.

16. DOR VII-10. I don't know what to do with this comment. Appendix VII was put in for what it was worth to attempt to give some insight. Its most important insight is the concept that the individual sequences chosen, for example, in the Status Reports or in Section 3.1 must not be considered in isolation as to their probabilities, but only as representative of many possible event sequences whose probabilities add up to give the curve of cumulative probability. Since Appendix VII helped me, maybe it will help the reader even though imperfect.

WM Moore Hanauer Easterling ? We should ... when Bob is here.

17. Easterling. Later.

Stephen H. Hanauer
Technical Advisor to
Executive Director for Operations

cc: H. Ornstein
T. Novak
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