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REACTOR SAFETY STUDY

The enclosed report constitutes the comments of the AEC Regulatory staff on the AEC Reactor Safety Study as presented in the WASH-1400 draft dated August 1974, distributed for comments on August 20, 1974. The report was developed by a Task Force appointed to conduct a technical review of the Reactor Safety Study. To augment the Task Force in certain areas, consultation with experts not on the Task Force and some not on the Regulatory staff was employed by some Task Force members and groups. The Task Force has also held discussions with the Reactor Safety Study staff and with Study contractors and consultants under the aegis of the Study staff in order to enhance our understanding of the work performed in the Study.

The present report is intended to be only a beginning. The Regulatory staff is here attempting only a technical evaluation of the Study Report dated August 1974. In accordance with the Commission policy (39 FR 30964), the staff will take further steps in evaluating and applying the Study methods and results after interested parties have commented on the draft Study Report and a final report has been issued and evaluated.

Our conclusions are summarized in the first section of the review. We believe that the Study represents a significant breakthrough in the quantitative evaluation of the risk to the public from nuclear power plants. We believe the Study was generally too conservative in the assessment of accident probabilities. However, the Study did not address all accident probabilities as we think they should have and we have identified some that might contribute significantly. We believe the Study was optimistic in some aspects of consequence calculation. Considering all of these together, we find ourselves in substantial agreement with the risk conclusions of the Study; that this risk is very low and on a comparative basis is smaller than most other risks to which we are exposed.

We have commented at some length on various aspects of the calculations used to arrive at this result. Our comments also include a number of suggestions aimed at improving the Report's explanations and justifications of the methods and quantitative results.

Members of the Task Force have also developed some more detailed comments and questions that we would be glad to discuss with the Study staff.

L. Manning Muntzing
Director of Regulation

Enclosure:
"Review of the Reactor
Safety Study"

REVIEW OF THE

REACTOR SAFETY STUDY

(WASH-1400)

DRAFT OF AUGUST 1974

Comments by the
AEC Regulatory Staff

November 1974

U. S. Atomic Energy Commission
Washington, D. C. 20545

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1. OVERVIEW AND SUMMARY

We believe that the Study represents a significant breakthrough in the quantitative evaluation of the risk to the public from nuclear power plants. This work is by far the most comprehensive, systematic, quantitative effort yet conducted in this field. It provides information of a new dimension to assist in making informed decisions when the risk is a significant consideration. It is therefore an important step in the evolution of safety technology.

Our evaluation of the Study in this review is confined to a technical assessment of the Study Report Draft of August 1974. Further action in evaluating and applying the Study methods and results are deferred for later consideration after a final Safety Study report has been issued and considered.

Two principal conclusions concerning nuclear risks are made or implied by the Study: First, that the risks of nuclear power are very low and, second, that these nuclear risks are lower than many other non-nuclear risks to which we may be exposed during our lifetimes. We believe that these conclusions are correct, although we do not wish to prejudge how a revision based on our comments and those of other organizations and individuals could affect the computed results of this complex study.

The comparison of nuclear and non-nuclear risks is a useful yardstick to calibrate the reader's understanding of the probability results, but the treatment of risks should be more consistent regarding onsite effects.

The calculation of "individual risks" from nuclear power plants and other causes is necessarily an averaging process. The Study results in this area would be more precise if they included as a refinement the large variability of such risks to the individual arising from, for example, differences in proximity to nuclear plants or differences in proximity to dams. The comparison of societal risks is a complex process, on which there can be differing interpretations. The Study report's comparison of risks is one such effort and an important contribution but is not necessarily the final judgment on the matter.

A commendable feature of the Study is the objective to be "realistic", to avoid overoptimism or overpessimism in the calculations. The use of frequency distributions for taking uncertainties into account explicitly is an important contribution. Also valuable, in our opinion, was the Study's attempt to treat core meltdown and containment failure modes in a realistic way considering the limitations of present knowledge. Some quite conservative (overpessimistic) assumptions were made, however, regarding the frequency of several initiating events and the criteria for the successful operation of several engineering safety features. Other assumptions, whose realism is difficult to evaluate, were necessarily made in core meltdown and containment failure scenarios, where the available technology base is not as firm as in other parts of the calculations. As additional information becomes available in the future, it should be possible to make a more detailed and quantitative evaluation of how realistic the calculated risks are. Specifically, we believe that the frequency of core meltdown given in the Study is substantially higher than reality, as are the frequencies given for many of the initiating events and the probabilities given for some of the system failures.

The explicit inclusion of human error in the fault trees is an important improvement over previous evaluations, as is the comprehensive and detailed consideration given to common mode failures in all phases of the calculations. The latter would be improved, however, by explicit inclusion of related failures attributable to design and manufacturing errors, over and above the "failure-rate coupling" (with individual failures remaining independent) now included.

An important and valuable feature of the Study is the use of event trees to keep track of the probabilities and releases of accident sequences and to direct attention to dependencies and potential for common mode failures. The development and evaluation of dominant sequences is valuable. It is noteworthy that the large cold-leg loss-of-coolant accident and the sudden reactor vessel rupture, subjects of much recent controversy, are shown by the use of these methods not to be dominant events.

It is the goal of such calculational techniques to consider all paths, with different probabilities and consequences, leading to significant risk. We believe that some initiating events have been excluded without adequate justification in the Study report, for example, an earthquake more severe than the Safe Shutdown Earthquake.

In this connection, more information on determining the degree of sensitivity of the results to the various factors would be valuable. In various places, the Study report suggests that one quantity or another need not be estimated accurately because its inaccuracy would not change the result significantly. A systematic discussion of which quantities are important and which, if altered, would change the results, would be helpful. In addition, the potential effect of a combination of several minor differences, all in one direction, should be addressed. The Study report implies that no single variable could affect the results significantly, but does not present an evaluation of the effect of such combinations.

The probability and consequences of the release of significant amounts of non-volatile material to the environment during postulated disruptive events have not been adequately addressed. The results of alternative health effects assumptions and the effect on the results of inclusion of the cost of illnesses should be more thoroughly presented.

The Study report would be greatly improved by including more facts - equations, numbers, code listings - to go with the narrative descriptions. Consistent and accurate terminology should be employed, and a complete glossary given. Approximations and assumptions used should be stated, justified, and examined for possible bias introduced into the result. We believe that the Report can and should be improved to communicate fully and convincingly the methods and results of this valuable work.

2. METHODOLOGY OF THE STUDY

2.1 GENERAL OVERVIEW OF THE METHODOLOGY

This section of comments discusses general conclusions (with some examples of the bases) regarding the methodology of the Study. The general opinion of the Task Force reviewers is that the Study represents a significant breakthrough in the quantitative evaluation of risk from nuclear power plant operation. This work represents the most comprehensive, systematic, quantitative effort yet conducted. The judgment of the reviewers is that the methodology is generally sound. We have, however, a number of detailed comments.

2.2 RESERVATIONS CONCERNING THE METHODOLOGY

We have two basic reservations about this type of endeavor. One is that even though the calculated result appears defensible, it is only as good as human imagination has permitted (in the development of models and logical constructs) in the absence of statistically significant experience. A more systematic discussion should be given of the spectrum of possible initiating events and the criteria for choosing those to be included in the calculations. In Section 3, we discuss some examples of initiating events whose omission has not, in our opinion, been adequately justified.

The other reservation is whether the various factors in the risk assessment are independent. This is taken into account in detail in the Study for initiating events and safety function availabilities, and discussed for the other factors. Future additional study of possible additional interactions seems warranted.

2.3 TREATMENT OF UNCERTAINTIES

Although the Study includes detailed calculations of the uncertainties in accident sequence probabilities, and includes distributions of weather and population, the risk results are published with an assertion regarding an estimate of their uncertainty, without any calculational basis being given. We recognize the limited information available in some of these areas but hope that future work will allow more complete quantitative derivation of uncertainties in the results. In particular, the calculations of core meltdown and release fractions, human effects, and property damage include little attempt at recognizing and accounting systematically for uncertainty. The origin of the risk uncertainties given should be explained.

The use of 5 and 95 percentiles to characterize the spread of a distribution is clear enough, but these are not "peak" values in the sense of being limits or bounds, as sometimes implied. The use of "peak" in Section 5.7 of the Report is applied to two different values: the bounds of WASH-740 and the combination of the most severe categories of releases, weather, and population considered in the Study. The Report should discuss the comparability of these measures.

2.4 USE OF APPROXIMATIONS, ASSUMPTIONS, AND METHODS

The Study involves a number of significant approximations. All models are by nature approximations. Of necessity, the calculational procedures involve many arbitrary features and correction factors to reduce the project to a workable size. The report makes it difficult for the reader to determine the origins of some of these features, and the justification for some of the approximations.

In a number of places, approximations are justified on the basis that the results (the risks, presumably) are not significantly altered by any inaccuracy thus introduced. The reader needs a discussion of which items are sensitive and would alter the results if uncertain or incorrect. As the Study warns, this must be done very carefully if any of the numbers calculated in the Study are to be used for any purpose other than overall risk evaluation.

2.4.1 LOG-NORMAL DISTRIBUTION

The log-normal is an intuitively satisfying distribution to many people for representing uncertainties, and is used throughout much of the report. The use of the distribution, however, should be discussed more comprehensively in the text and should be based as much as practicable on reliability data.

2.4.2 MEDIAN AND "SMOOTHING"

We believe that the use of the median instead of the mean of the accident sequence probability distributions introduces an error of about a factor of two in the optimistic direction (the mean is larger than the median). This should be evaluated and discussed in the Report.

The "smoothing" of the release categories (Section 5.3) should be discussed in more detail. It would have been preferable to carry uncertainties in release magnitude along with the computation of consequences of each sequence instead of attempting to compensate for it afterwards by adjusting probabilities on the basis of the sequences which have been placed in the adjacent release categories. The validity of this procedure depends on a showing that the variability of releases is covered by the categories included in the smoothing. This subject should be discussed, since in some cases, the contribution from smoothing into adjacent categories dominates the probability contribution from the sequences within a particular release category.

2.4.3 AVERAGING

Some mathematical expressions of the methods of averaging, such as in the case of population distributions, along with examples, would be helpful in understanding and using the report. The use of the term "peak" in contrast to "average" may be taken to mean "maximum" when in fact certain peak values appear to have some inherent averaging. Some insight might be gained by showing the range of values on curves such as Figure VI-3 of the Study report.

2.4.4 CATEGORIZING SEQUENCES

The calculation of releases was performed only for large LOCA accident sequences. No basis and little explanation is given regarding the fitting of all other sequences into these categories. In view of the fact that initiating events other than the large LOCA dominate the probabilities of all core melt release categories, it would seem to us to be important to discuss the release characteristics of these dominant sequences.

2.4.5 OTHER APPROXIMATIONS

Some other mathematical and calculational approximations also seem not to be well discussed and justified in the Study Report. The correct formulas for probability elements should be given and then followed by approximations if their use is desirable and their accuracy is justified. For example, formula (3), page 160, Appendix II, Vol. 1, could be described, if appropriate, as "the incomplete gamma function for which the infinite series expansion is . . . . In this study, sufficient accuracy is attained by omission of all terms except the first." Sometimes the approximations give values less than the exact and other times they give values greater than the exact. There is nothing in the report to convince the reader, or the independent reviewer, that all the approximations and omissions do not in some instances produce accumulations that are significant.

2.5 FAILURE RATE DATA

In the Study it is acknowledged that one of the weak links in the analysis was the imprecise data base. While the imprecise data base may not have a major effect on the overall risk evaluation, its impact on intermediate probability statements may be more important. Accordingly, the method, iterations, and rationale employed in arriving at the data base used should be discussed.

More specifically, it would appear that an uncertainty may be introduced by applying some component failure data (Section 4.2.2.2) obtained from non-nuclear applications to nuclear applications. The nuclear application may produce greater (or unique) stresses, or present a more hostile environment for the components, or conversely imply a higher order of quality and inspection. It is not clear that the uncertainties expressed in the Study report reflect those inherent in the method in which the data were used.

2.6 EXTRAPOLATIONS TO 100 PLANTS

The Study calculates the risks per reactor year using one BWR and one PWR as prototypes, extrapolates the results to the 100 nuclear reactors already approved, at least for construction, and expected to be operating by about 1980, and compares these extrapolated risks to certain risks faced by society and individuals from a variety of sources to conclude that nuclear risks are relatively small.

The use of Surry 1 and Peach Bottom 2 as prototypes is valid to the degree that these plants are typical. It is certainly true that all these plants were built in the framework of U.S. industry design and USAEC review. The "rules" changed a lot from the first plant's design in the late 1950's to the hundredth plant's design in the early 1970's. Even though the approach to many of the "rules" was similar throughout this period, some consideration should be given to this problem. But the Study does not include the necessary information and consideration to allow determination of the degree to which these plants are typical. It is true that all plants are different; the differences are matters of degree. It is also true that only detailed study of a plant can form the basis for risk determination of the depth and scope of this Study. A list of the major non-typical features of Surry 1 and Peach Bottom 2 should be compiled and some evaluation made of the significance and bias, if any, introduced by the non-typical features of these and other plants.

During the course of our review, we identified certain atypical features for the two plants evaluated. For the PWR these include a subatmospheric containment system, an elevated intake canal, the piping arrangement involved in the "V Sequence," and the normal feeding of emergency electrical buses from offsite power sources. For the BWR they include the design of the rod sequence control system and the normal feeding of emergency electrical buses from offsite power sources.

3. EVENT/FAULT TREES AND ACCIDENT SEQUENCES

3.1 GENERAL OVERVIEW

We believe that the event tree/fault tree approach used is well founded on established mathematical and engineering principles, is acceptable, and is a useful approach for assessing the probability of core melt events.

3.2 EVENT TREES

The Task Force believes that the Study considered the vast majority of significant events in an appropriate and quantitative manner. However, it appears to us that some events that may be significant have not been considered. Some other events, which were considered but set aside as unlikely, appear to have been assigned too low a probability. With but two exceptions, we do not believe that inclusion of all of these items which we identified would potentially alter the results of the Study. The two exceptions are the severe seismic event for all reactors and the rod ejection accident in the BWR.

3.2.1 REALISM AND CONSERVATISM

Many of the procedures and assumptions employed in the development and use of event trees in the Study contribute to an overall conservatism in the prediction of core melt. This general subject is discussed in several instances elsewhere in our comments. However, because of its basic nature to the event tree formulation, the conservatism inherent in the "go-no go" philosophy for both initiating events and system performance is discussed at this point.

Most failures are partial in nature; for example, a pipe is much more likely to endure a partial rupture than a complete failure, the loss of net positive suction head is more likely to result in partial or temporary failure of a pump than in its total loss, and the failure to switch to hot leg ECCS injection for a PWR one day after a LOCA will more likely have no serious impact on core cooling rather than result in total failure of the emergency coolant recirculation function. However, we fully recognize the difficulty of analyzing the effects of partial performance or failures, such as of engineered safety features, and feel that the Study was practical though conservative in its approach.

3.2.2 SEISMIC EVENTS

The Study addresses the probability for the occurrence of severe seismic events and the probability for the failure of systems should such an event occur. The position stated in the Study report is that the probability for the occurrence of an earthquake in excess of a typical Safe Shutdown Earthquake (SSE) defined by the Regulatory staff is in the range of 10^{-4} to 10^{-6} per year. For the SSE, the Study estimates the probability for the attendant failure of systems sufficient to result in core meltdown to be in the range of 0.1 to 0.01. On this basis it was concluded that seismic events would not contribute significantly to the probability for core meltdown and no specific assessment of seismically-initiated events were included in the Study. In our opinion the treatment given to seismic events in the Study is too shallow to justify the conclusions reached. We believe the assessment of the potential for and the consequences of large earthquakes should be reevaluated, and that the contribution to the risk from large earthquakes may be relatively significant when other dominant events are considered on a more realistic basis.

More specifically, we believe that the magnitude of seismic events worthy of consideration should cover a range of intensity of events. The lower end of the range should include those events of the least intensity, possibly Modified Mercalli VI, which could result (with low probability, taken into account as elsewhere in the Study) in damage to non-Seismic Category I structures, either on site or off site, thereby possibly initiating a transient. The upper end of the range should be greater than the SSE. The probability of failure of a component varies with the intensity of the stresses to which it is subjected. To the degree practical this approach should be developed and factored into the final report.

The discussion of earthquake probability is weak because it fails to properly address (1) the variation of earthquake risk across the nation, (2) the nature of error and uncertainty in the data available from the historic record, (3) the statistical models appropriate for estimating earthquake recurrence, and (4) the mean area affected as a function of intensity. It appears that use of a more appropriate relationship between intensity and acceleration would result in a higher computed probability of damaging accelerations arising from a large earthquake by a factor of about five.

There are at least three problems relating to response of components and structures to earthquake accelerations which were not adequately addressed. First, the probability of small earthquakes causing multiple failures in non-Class I systems might affect the probability of core melt by creating a transient condition in which more than one normal system would be unavailable. This might increase the probability of a challenge to the seismic Class I safety systems. Second the evaluation of the probability of failure of vital systems in an SSE did not include consideration of the systems which were not assessed in Appendix X because sufficient information was not available. Half of these were assumed to fail. But these failures may not be independent of each other. It is worthy of consideration that if a system is improperly designed, its redundant system may also be improperly designed. The probability of core melt in an SSE under such circumstances must include consideration of this possibility. On the other hand, if all systems are properly designed against an SSE, we believe that the probability of core melt as a consequence of an SSE could be justified to be 0.01 or lower on the basis of margins available in the structures.

Consideration should also be given to human error, difficulty of evacuation or of remaining sheltered in buildings, and other factors attributable to a severe earthquake.

Third, the probability of an earthquake significantly larger than the SSE has not been taken into account in the risk evaluation. Increasing the severity of the postulated earthquake acceleration at the site would decrease the probability of the initiating event but would increase the probability of consequent structural and system failure, though not necessarily in the same proportion. The Study includes neither the considerations of probabilities of such severe seismic events and of such failures, nor any investigation of failure modes so induced and whether the event trees and release categories derived from the large LOCA are adequate to describe these different events.

3.2.3 ROD EJECTION ACCIDENT

The rod ejection accident in a BWR is addressed in the Study. On page 197 in Appendix I it is stated that the reactor has been designed so as to make the likelihood of a rod ejection accident negligibly small. The justification for this position is not provided. We believe that the probability for a severe rod ejection accident is indeed small, but that it should be evaluated quantitatively and is on the order of that for a number of other initiating events that have been quantitatively assessed in the Study. A rod could be in an ejectable state if the rod drive housing fails. This housing is essentially a pipe about six inches in diameter. The probability for rupture of such a component should be assessed; it could be the same as for a small LOCA event of the same break area, which the Study estimates to be 3×10^{-3} per reactor year. The ejection of the rod from the core is designed to be restricted by a control rod drive housing support structure which is permitted to be removed when the reactor is shutdown. The removal feature is necessary for maintenance and inspection purposes. The support structure is not amenable to testing and thus its ability to perform its design function depends upon the correctness of the design, fabrication, and installation. It is therefore susceptible to human error in that it may not be reinstalled or may be installed incorrectly or imcompletely following its disassembly. A probability on the order of 10^{-4} for the failure of the structure to perform its design function would lead to an overall probability of about 10^{-7} for the rod ejection accident. In view of the severe potential consequences of such an accident, which could exceed those of some of the other accidents considered, such a probability level would perhaps be a dominant event.

3.2.4 OTHER EVENTS

We believe that some other events, both internal and external to the plant, should be assessed in greater depth than presently in the report. These include the potential effect of control room uninhabitability, fire, high trajectory turbine missiles, aircraft impact fire effects, core flow blockage, failure of containment internal structures, vapor suppression failures in BWRs that lead to a core melt, and BWR main steam line breaks. Several of these are touched upon in the report but their omission from the quantitative assessment is not justified in a way consistent with the rest of the report.

We believe that the discussions and analyses of addressing non-core accidents should be expanded and more thoroughly discussed.

3.3 FAULT TREES

The fault tree approach uses binary modeling of faults in that all components of a system under study are either in a "failed" or "unfailed" state. While we agree that binary modeling is a characteristic feature of fault tree analysis; i.e., continuous spectra of faults cannot be treated, we believe intermediate states from some of the more critical components should be considered if practical. For example, incipient melting of the core can be divided into that melt which is realistically expected to be arrested and that which proceeds to reactor vessel and containment failure. Valves and pipes can be partially failed without losing their ability to perform their function creditably. Failure of the reactor protection system is defined as the failure of two or three neighboring rods to insert when required. For several of the accident sequences considered, this degree of partial failure can be corrected prior to the occurrence of any major core melt.

3.3.1 TRANSFERRED SUB-TREES

A number of the fault trees include the use of transfer symbols which indicate use of previously developed results from sub-trees. Whenever the results of sub-trees are transferred into system fault trees, and whenever system fault tree results are transferred into accident and transient event trees, the assumptions used to develop the sub-trees and system fault trees should be examined to assure that they are consistent with the event tree sequence under consideration. An example of this concern is the development of event C (RPS failure) for the large LOCA in BWRs. The sub-tree of RPS failure indicated a median failure probability of 1.3×10^{-5} per reactor year, correctly assuming no credit for operator actuation of the manual scram system. When the result of this sub-tree was used in analysis of the transient sequences TC-..., the assumptions should have been modified to allow credit for manual actuation of the scram system, because for some transients there will be sufficient time for reliance on operator action. This should reduce the probability for some TC-... sequences substantially. Another example is the use of the short-term (half-hour) rather than long-term (25 hours) probability for HPSWS failure in Event W on page 73 of Appendix V. We recommend a reexamination of the development assumptions whenever pre-analyzed sub-trees are used in different system fault trees.

3.4 SABOTAGE

The Safety Study report (Section 7.4.2) states that "any consequence produced by sabotage could not exceed the largest predicted by the Study". We believe the Study report should discuss the basis of this conclusion in greater detail.

4. PROBABILITIES

We have reviewed the probability estimates, calculations, and supporting failure rate data presented in the Study report. Most of these agree with our understanding of the failure rate data. A few items seem worth commenting on here as we believe discussion of these items in the final Study report is desirable.

4.1 INITIATING EVENT PROBABILITIES

It is our opinion that the probabilities selected for the large LOCA event, the small LOCA event, and the small-small LOCA event are overly conservative by a factor of at least ten and perhaps as much as one hundred. We also believe that the probabilities of the other assumed initiating events, with the exception of the transient events, are also conservative to a similar extent. The probabilities for the transient events appear more reasonable.

Accordingly, we recommend careful and more rigorous assessment of all available information in order to arrive at more "realistic" values for the probabilities. The most significant of these are discussed in the following.

4.1.1 PIPING FAILURES

The failure probabilities set for piping in the Study are based on historical data on pipe failures, as obtained from the experiences of other segments of industry. Because these data were obtained from experiences sustained in plants which include petroleum refineries, chemical processing plants, and electric utilities, the piping systems data were not entirely representative of nuclear power plant piping in terms of code design requirements, non-destructive examination of weld joints, piping configurations and system extent, quality assurance measures, and the degree of inservice inspections. These factors should contribute significantly to the quality level built into nuclear piping systems over those in non-nuclear systems, and should decrease the failure probabilities expected during service. A lowering of the probability of failure estimate is therefore to be expected when the superior quality factors associated with nuclear power piping are taken into account, together with the diverse means of detection of incipient failures. In our opinion, the probability of failure (severance) of large nuclear pipe should be reduced by an order of magnitude, which yields an estimated occurrence rate of about 10^{-5} per plant year. The probability of severance of small pipe in nuclear service should also be lower, although perhaps not by a full order of magnitude.

As a further example of the conservatism that appears to have been used in the selection of failure rates for LOCA-initiating events, it was assumed that 5% of all piping in a plant, or about 8500 feet of piping, is large LOCA sensitive; i.e., could lead to a large loss-of-coolant accident, if ruptured. This assumption is very conservative.

4.1.2 PRESSURE VESSEL FAILURES

The estimate of pressure-vessel failure probability used in the Study was based on recent evaluations by the ACRS and the Regulatory staff. But the failure rate values chosen by the Study were those derived by the ACRS and the Regulatory staff as conservatively high for reactor vessels. It is true that the Study estimated that even higher values would not affect the calculated risks, but this does not justify labelling such values as "realistic".

4.1.3 PWR "CHECK VALVE" SEQUENCE

The Study contains extensive discussion and analysis of a potential PWR accident sequence associated with the arrangement of check valves isolating the LPIS from the RCS. The calculated probability for this "Event V" is given as 4×10^{-6} per reactor operating year. We believe that the assessment of Event V is overly conservative in three basic respects, namely (a) the failure data used for the check valves, (b) the assumption that LPIS rupture was certain to occur given a check valve failure, and (c) the assumption of certain failure of the remaining systems to cool the core adequately.

The analysis was calculated using failure rates for the likelihood of one of these check valves failing open (λ_1) and then having the other check valve fail by disintegration of the disk permitting massive backflow (λ_1). These failure rates are taken as:

$\lambda_1 = 2.6 \times 10^{-3}$/year, and $\lambda_2 = 8.8 \times 10^{-5}$/year.

Tracing these rates to Table III-1 in Appendix III, we find that 2.6×10^{-3} is the failure rate of valve leakage, not failing open. Since no failure data for failing open were available, the Study conservatively used the failure rate for valve leakage. We believe that even heavy leakage could be tolerated without LPIS rupture and heavy leakage is not descriptive of the large LOCA which needs the flow delivery of the LPIS. The conservatism in this failure rate may easily be one or two orders of magnitude in probability.

Similarly, failure data for internal (disk) rupture were not available, so the probability of valve body rupture was used for λ_2. We believe that this is conservative.

An external rupture of either check valve would take place inside containment and would disable the LPIS only in that it would have to be realigned to feed to the hot legs of the RCS to deliver cooling water.

A further conservatism is contained in the assumption of certain LPIS rupture. The LPIS is designed to 600 psi by piping codes which include significant design margins. The blowdown of the RCS into the LPIS tends to reduce the driving pressure and the path of flow entails additional pressure drop. In addition, the piping between the high pressure piping and the next check valves upstream (at the pump discharges) is equipped with three small relief valves, RV 1845 A, -B, and -C. In view of all these factors, it is not certain that the LPIS will rupture in Event V.

Lastly, it is assumed that Event V causes the total loss of the LPIS leaving the core inadequately cooled and certain to melt. There is, of course, a very significant probability that the accumulators and HPIS pumps together may adequately cool the core.

In summary, Event V has been analyzed very conservatively leading to a median estimate of 4×10^{-6} per year for the probability of core melt due to the event. A more detailed and realistic analysis would reduce this probability estimate substantially.

4.2 EFFECTS ON FAILURE RATE DATA

Table III-1 of Appendix III presents the failure rate data used in the Study. Other tables in Appendix III present comparisons of non-nuclear and nuclear experience data as well as associated error limits. We believe that the basis for selection of failure rate data should be more thoroughly discussed particularly because the tables of Appendix III show some apparent discrepancies. Further, it appears that the methods used for establishing failure rates and error intervals, and for rounding off may have introduced errors in the results.

4.2.1 TESTS AND MAINTENANCE

There is a conservative cast to the treatment of Test and Maintenance with respect to system unavailability in the Study report. Testing and maintenance programs are essential to achieving high reliability of these systems. Ideally, systems should be designed so they do not have to be made unavailable during testing. The systems evaluated are not ideal in this respect. The necessary testing and maintenance, therefore, contribute to system unavailability. The evaluation includes the repeated assumption that scheduled maintenance is performed while the plant is in operation. We believe it is typical to perform only the minimum required maintenance on ESFs when the plant is operating. This is particularly true with respect to standby equipment, such as spray pumps, LPIS pumps, etc. For high use items like the charging pumps, an installed spare provides the extra maintenance availability needed without excessive risk of plant shutdown for lack of a full complement of ESFs.

An example of this conservatism is as follows. For the BWR plant, event U is defined as the availability of the HPCI or RCIC systems for makeup inventory. In determining the failure probability for HPCI and RCIC to provide makeup water, very high unavailability factors are applied to both HPCI and RCIC. A major fraction of these unavailability factors is assigned to account for the periods when these systems are tagged out of service for performing maintenance on valves and other equipment. The simple treatment shown for combining the unavailabiulity factors for HPCI and RCIC seems to allow both HPCI and RCIC to be out of service for maintenance at the same time. This situation is not normally allowed by the technical specifications. Correction of this problem should reduce the failure probability of both HPCI and RCIC by about a factor of six.

Moreover, we believe that the actual fraction of a year that either HPCI or RCIC is tagged out of service for maintenance is substantially less than that used in the Study. A cursory study of abnormal occurrence reports filed for 10 BWR plants representing 17 reactor years of operating experience suggests that the maintenance downtime should be about a factor of 8 lower than that used in the Study.

We recommend that a more extensive search be made of station operating logs and associated maintenance logs at a number of BWR plants so that more reliable values can be assigned to HPCI and RCIC unavailabilities due to maintenance. In addition, consideration should be given to reevaluating tests and maintenance unavailability in other systems where this factor is important.

4.3 DESIGN ADEQUACY

The impact of the results of Appendix X on the results of the report does not accomplish the goal expressed in the introduction to Appendix X. The statement is made that concern over external causes and severe environment resulted in this study and that "This design adequacy assessment was done to determine whether (1) the phenomena represent a common mode failure potential and (2) if dependencies do exist, they are factored into the probability predictions".

One might expect to find that the information developed in this portion of the Study was used to adjust the values of failure rates obtained for non-nuclear components in normal service. This is not the case however. There is no indication that these data were incorporated in any quantitative way into Appendix III or Appendix IV.

The use in Appendix X of averaging seismic adequacy across classes of component groups appears to be a less rigorous treatment in comparison to the methods used in the remainder of the Study.

Fire in some areas of the plant such as a cable room or diesel building might have a potential for inducing significant common mode failures. It is not apparent that this has been considered. Some discussion of the effects of fire is merited in the final report.

Material selection and possible degradation is an area that must be examined and discussed even if it is evaluated as insignificant or negligible later in the Study. Material selection, as a significant aspect of design adequacy, could be an important contributor to common mode failures or may result in early wear out and negate the value of random failure data used for the plant lifetime. If they are not to be considered, the basis for omission of such time dependent aspects of potential common mode failures should be discussed in Appendix X of the final report.

The data base, governing criteria, method of evaluation, and conclusions for the various components examined should be presented in a more orderly and consistent manner.

4.4 COMMON MODE FAILURES

The common mode failure (CMF) calculations do not include a correct treatment of causally related failures of identical redundant components due to design and manufacturing errors. The "failure rate coupling" that is included considers only successive or concurrent random failures whose rates are coupled by manufacturing or environmental effects. But non-random multiple failures of this type have actually been experienced in high-reliability systems such as reactor scram systems for which the calculated unavailability is believed by us to be lower than reality because these failure modes were not included in the calculation. On the other hand, for systems whose unavailability is dominated by single failures, these CMFs would not be expected to be important.

While the above comment represents our principal concern, we have also concluded that some of the common mode failure contributions from human error that are quantified in the Study have been determined in an overly conservative manner. As one example, the point estimate for the common mode contribution to unavailability of the PWR Auxiliary Feedwater System is 3×10^{-5} . This estimate is stated to be based on operator errors which result in the two valves in each of three parallel subsystem (two subsystems are associated with two electric motor-driven pumps and one with a steam turbine-driven pump) being left in the closed position after the required monthly tests. The probability for leaving one set of two valves in the closed position is given as 10^{-2}. A factor of 1/3 is assumed for detection of the error during walk-around inspections. The tests are done on the three parallel paths sequentially and on this basis some unspecified coupling is assumed to give an overall probability that all six valves will be closed of 3×10^{-5}.

The technical specifications require each pump to be tested on a monthly basis. The probability that the monthly tests of the pump and its two associated valves would be performed so that the two manual valves at the pump are left in the closed position should be low. The value of 10^{-2} assumed for this probability seems conservative but accepting that value, and a similarly conservative 10^{-1} value for the adjacent electric motor-driven pump parallel subsystem, results in a 10^{-3} value for the combination of the two subsystems.

The steam turbine-driven pump subsystem differs from the other two subsystems in design, appearance, and test procedures. Also, the test requires steam line valves to be opened and closed. These differences should logically lead to a failure probability nearer to the value for the initially tested electric-driven pump subsystem than to the value for the second such subsystem. A value of 10^{-5} for the overall system would result.

The factor of 1/3 assumed for detection of the facility valve status appears conservatively low. However, even assuming this value, the final system unavailability would be 3×10^{-6}. Even this value should be reduced significantly since the failure of the system to perform after a small LOCA would be readily detected and timely corrective action could be taken. While no credit has been assumed in this instance for the corrective actions and procedures which will accrue as a result of experience with operator errors occurring during normal operation, this type of evaluation is indicative, in part, of the basis for our conclusion that the point estimate values for common mode failures of this sort are overly conservative by an order of magnitdue or so.

5. CONSEQUENCES

5.1 GENERAL OVERVIEW

The consequence calculations in the Study are based on single-value estimates with a limited number of indications of sensitivity of the results to variations in individual parameters. The treatment of core melt, containment failure modes, radioactive material releases and health effects included a state-of-the-art review which, while usually adequate in its assessment of currently available information, was then used as the basis for deducing the behavior of molten masses and radioactive materials in environments for which the state-of-the-art studies acknowledged significant deficiencies in information. For example, dispersal of significant amounts of non-volatiles like La and Pu should be discussed; we think the Study report should show convincingly either that such dispersal is impossible or, more likely, that the probability of significant dispersal is very low even though the consequences could lie outside the bounds of the existing Study release categories.

We believe that the consequence section of the report deserves a good deal of additional attention both to present more adequately what was actually done in the Study and to treat explicitly the limitations and uncertainties involved in this part of the Study.

5.2 CORE MELT SEQUENCES

The calculation of core melting, melt-through of the reactor vessel and failure of the containment includes both items treated optimistically and items treated pessimistically. We recognize the limitations of knowledge available in these areas. Where uncertainties exist associated with the phenomena of heatup and meltdown of the core, and its subsequent movement and reactions, estimates were generally used which would in some cases cause the evaluation to deviate considerably from a realistic assessment. The final report should summarize the areas where non-realistic models and calculations are used together with an assessment of the extent of departure from realism and of the uncertainties in the results where this is possible.

The report describes the mean probabilities of containment failure and corresponding error bands for all containment failure modes except for "loss of isolation" which was developed in Appendix II. However, it is not clear that the probabilities described in Appendix VIII are, in fact, supported by the analyses in sub-appendices A through E. It would be helpful to provide further support for the values chosen and to present a sensitivity analysis for the range of the effects on consequences arising from the uncertainties in containment failure mode probabilities. Local containment failures should also be given further consideration. Additional clarification of the initial conditions, assumptions, program options, and meltdown scenarios used in the code used to follow the meltdown sequence (BOIL) and additional parametric studies are required before the adequacy of the code can be evaluated.

With respect to vessel melt-through, we suggest that the melt-through could occur in some fraction of cases in as short a time as 15 minutes following the dropping of the molten core into the reactor vessel lower plenum as opposed to the 60 minutes estimated in the Study. We believe that it is likely that the molten core would displace the water in the bottom of the vessel and that heatup of the steel would occur concurrently with heatup of the water, and that the heat fluxes from melt to vessel have been underestimated.

As discussed in Appendix VIII, the phenomena of the steam explosion event are complex and not well supported by experimental results. The numerical probability assigned to containment failure from this event is therefore necessarily somewhat arbitrary, and we agree with the Study that it may be on the conservative side. Possible early failures of the reactor vessel, which could change the timing and mode of containment failure, should be given further consideration in this regard.

5.3 RADIOACTIVE MATERIAL RELEASE

The values selected for the fractional release of radioactive material from the core region for the less severe accident sequences appear to be reasonable based upon the data and rationale presented in Appendix VII. A considerable uncertainty is acknowledged and indicated in the presentation of experimental data. However, as indicated in the Study, additional uncertainties are introduced by applying data obtained from short-term melt experiments with trace-to-moderate burnup in small fuel samples to long-term high temperature conditions and high burnup fuel in full cores, and these uncertainties should be resolved in future research activities.

The least well supported group fractional release of radioactive material is for the groups described as being of low volatility, and in particular with regard to the isotopes referred to as the lanthanum group.

The uncertainties of a factor of plus or minus ten in the half-time for release of the less volatile isotopes (which covers a range from 3 to 300 minutes during the vaporization release phase) and the factor of plus or minus 5 for the quantities of less volatile elements and compounds released should be considered in the sensitivity studies. The release times can be important with respect to containment failure modes and the effectiveness of evacuation.

While an "oxidation release component" is assumed to result from a violent steam explosion, no consideration is given to formation of an aerosol that includes low volatility materials such as plutonium oxide, lanthanum and strontium. This potential airborne source is not treated although the statement is made that a steam explosion event would result in the scattering of finely divided UO_2 (containing radioactive material) into the atmosphere outside the containment or into the air-steam atmosphere inside containment. In this regard, it should also be noted that the postulated low release fraction during vaporization for "refractory oxides" was partly predicated on recondensation of dense aerosol clouds on surfaces. A steam explosion might cause liftoff of condensed particles from surfaces into the containment atmosphere. Any material entrained in the water which would be ejected during this violent episode also would be a likely source of aerosol particles. Contaminated water release from the containment to the ground surface and thence to water supplies seems also to have been neglected.

The discussion of methyl iodide formation under core melt conditions uses a reference developed in part by the Regulatory staff which was directed to the determination of maximum methyl iodide formation under loss-of-coolant accident conditions with a postulated spectrum of releases for the volatile iodines. The reference did not address the question of the environment which might exist in the containment during a gross melt sequence. Indeed, the experimental information used in the reference appears to support the thesis that methyl iodide formation in the containment is dependent on the absolute amount of free organic material in the atmosphere. In the environment associated with a core melt, substantial amounts of organic materials could be released into the containment atmosphere. The methyl iodide fractions may have been substantially underestimated for this situation.

In the same way, removal rates for both gaseous and aerosol radioactive material by the engineered safety features are selected for evaluation in the Study from data obtained under environmental conditions which differ markedly from some of the accident conditions to which they are applied. While the selected values for removal rates used in the Study evaluations are within the range of possibilities, we believe the removal actually experienced could be appreciably less than presented in the Study, and suggest that this uncertainty should be included in the evaluation.

The following conclusions were reached based on our independent calculations and evaluation of the adequacy of the isotopes selected for the computation of consequences.

(1) For the fractional release value range given in Table 5.1, the use of the selected isotopes gives satisfactory predictions of long-term organ doses from inhalation or ingestion of the source, 30-day lung dose, and total body dose from cloud passage.

(2) We have suggested earlier in this section, the possibility of larger releases of non-volatile materials, such as La and Pu, in accidents of very low probability. For such releases, the Study's selected isotopes would not be adequate for estimating consequences. The elements in the Study's lanthanum group account for about 70% of the total internal dose potential in the core but the small release fractions result in a small contribution to the computed doses in the present release categories of the Study.

The selection of the characteristic time of release for the release categories needs additional explanation. For several release categories, the time selected appears to be significantly later than the bulk of the release. Selecting an earlier release time, albeit a somewhat smaller release fraction, could have a substantial effect on the degree of evacuation that can be accomplished and, thus, on the consequences of these release categories.

Additionally, the characteristic time of release could be significantly decreased if more ESF pumps than minimum were to operate initially, The use of Pasquill weather types (Section 4.3.1), the chosen wind speeds and fixed angular width seem reasonable considering the data available and the given source release duration. However, the assumption that a given weather type exists during the entire travel time over a distance of 500 miles represents a substantial overconservatism for the weather types with the poorer dispersion charactersitics, as recognized in the Study.

Consideration should be given to including, in the release model for containment melt-through, the effect of the layer of high-permeability material found in many designs under the containment floor and adjacent to the outside of the below-ground portion of the containment walls and the effects of containment uplift for the PWR.

5.5 EFFECTS OF RADIOACTIVE MATERIAL

The assumptions used in the Study's health effects model are apparently based on a selective use of the BEIR report. Consideration should be given to the uncertainties, reservations, and alternative assumptions that are part of the BEIR report. For example, a value of 100 latent deaths per 10^6 man-rem is used and indicated to be derived from the BEIR report. However, the body of the BEIR report contains an alternative model (relative risk model) from which somewhat larger numbers can be derived. In view of this, the Study statements that the selected risk values are "upper limits" should be qualified and the reasons for model selection given.

The risk values in the BEIR report are for thyroid cancers, not nodules. The Study appears to have used these risk values for thyroid nodules and states that there is a low probability of nodules yielding cancers. We also note that the current survival rate for patients with thyroid cancer is about 80 to 85%. The estimation of delayed health effects should include a separate calculation and tabulation of the estimated numbers of thyroid and other cancers resulting in death, and also non-fatal cancers. Separate consideration of thyroid nodule frequency appears to be of limited value in the total risk evaluation.

There are several areas where the uncertainties in health effects models as discussed in the BEIR report could substantially affect the consequence calculations and should be utilized in the presentation of results. The estimates of excess cancer mortality are characterized by the BEIR report as possibly in error by as much as a factor of five, depending on the assumptions made, as noted in the Study. With respect to somatic risks, a risk value twice that used by the Study could be derived from the BEIR report.

The Study selects values for predicting acute biological effects from doses to the lung (3000 - 5000 rads) and GI tract (2000 rads) based upon extrapolations which should be explained as well as the available information allows.

Coincidence of high specific organ doses and whole body doses might lead to a higher incidence of lethality than is apparently calculated from the separate effects. In particular, doses to bone and bone marrow during the 30-day integration period or over the 50-year period are not apparently included.

In view of the amount of excitement and activity which would be generated by the dissemination of news of the accident and the implementation of an evacuation plan, an inhalation rate of 3.5×10^{-4} m^3/sec would be more reasonable than the chosen average for a standard man. This would have a direct effect on the computed consequences.

In the discussion of the error spread on consequences, the sensitivity studies do not show the effect of varying the source term within the bounds placed on those parameters used to generate the source term. Although the sensitivity studies results indicate that changes in any one parameter caused changes in the consequences of no more than a factor of three, combinations of events may exceed the factor of three.

Additional consideration should be given to situations in which several factors are varied simultaneously, particularly for lethality calculations which are nonlinearly related to the dose.

The property damage model used in the Study does not include the cost of illnesses and other non-property costs. We believe that an assessment of non-property costs would be of interest. The property damage probability curves (Figure 5.9 and others) suggest that there are very few accidents in the range $\$10^6$-$\$10^8$. This is not consistent with the data in Table VI-20.

The comparison of nuclear and non-nuclear risks should be consistent in the treatment of human effects on station personnel and of damage to onsite property. Consideration should be given to comparing these risks both with and without onsite effects included.

The use of entire-population averages as estimates of individual risk is a valid basis of comparison only for those risks that are reasonably uniformly distributed over the population. Risks from dams, aircraft, and nuclear reactors, for example, are not uniformly distributed. The individual risk estimates would be much more valuable if the wide variation were acknowledged and some functional dependence developed.

We believe that the results should be presented in terms of a more realistic risk to an individual. This could be accomplished by showing the risk as a function of distance from, for example, a three-unit station. This would provide an additional perspective. This technique would also allow consideration of risk to individuals from multiple sites in a region.

In general, the frequency of low-probability non-nuclear events such as dam failures with high fatalities appears to have been overestimated. The uncertainty in these non-nuclear risk calculations should be indicated.

We believe that when the health effects to affected groups are treated equivalently in both nuclear and non-nuclear cases and when the uncertainties in release phenomena and health effects are taken into account in the nuclear case, the uncertainty band of the estimated risk from nuclear accidents may overlap the estimated risk from the lower probability non-nuclear events considered. In any event the risks will remain very low.

On the basis of our review of the Study's comparison of the Study results with those presented in WASH-740, we believe that the tabular comparison and associated text should reflect better the discussion of the release of non-volatile materials in Section 5.3 of this review.

The principal difference in the release terms in the two calculations is the release of substantial amounts of non-volatiles (referred to as the lanthanum group in the Study) in the worst WASH-740 release case (the "50% release" case). If a steam explosion which dispersed low-volatility elements could occur in conjunction with a containment failure or if the available low-volatility elements could otherwise be swept or released from the containment, the Study source term and consequences might resemble the "50% release" case in WASH-740. Even though this release mechanism is substantially lower in probability than those considered in the Study, an assessment of the probability and consequences of this category of event should be made to assure that the contribution to the risk from this event is not significant.

Another case presented in WASH-740, the "volatile release" case, is similar to the PWR-2 case of the Study for external doses. The WASH-740 consequence model is also not dissimilar to that used by the Study. An important difference in the two calculations is that the Study has quantitatively evaluated probabilities of certain consequences and thereby estimated risks, and WASH-740 did not. This points out the major value of the Study - the ability to treat both probabilities and consequences.

AEC REGULATORY STAFF EVALUATES REACTOR SAFETY STUDY

The Atomic Energy Commission's Regulatory Staff has concluded that the Reactor Safety Study prepared for the AEC (WASH-1400) "...represents a significant breakthrough in the quantitative evaluation of the risk to the public from nuclear power plants."

In a letter to Dr. Norman C. Rasmussen, who directed the two-year study, Director of Regulation L. Manning Muntzing also commented on the Study's conclusions in regard to these risks:

"...we find ourselves in substantial agreement with the risk conclusions of the Study; that this risk is very low and on a comparative basis is smaller than most other risks to which we are exposed."

The Regulatory Staff, as a result of its technical evaluation, further concluded that: (1) the Study generally overestimated the probabilities of accidents but did not address all accident probabilities, some of which might be significant; and (2) that the Study underestimated the consequences of some accidents.

Copies of the Regulatory Staff's comments on the Rasmussen Study are available for inspection in the AEC's Public Document Room at 1717 H Street, N.W., Washington, D.C.

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NUREG-0625

Report of the Siting Policy Task

Office of Nuclear Reactor Regulation

U.S. Nuclear Regulatory Commission

SEPTEMBER 1980

An Emerging Battleground

The nuclear industry, both in the U.S. and abroad, is preparing for a battle over remote reactor si ing rules being developed by the NRC which appear to curtail sharply the number of potential locations available for construction of new reactors. Some critics claim the rules, if adopted, would eliminate all presently feasible reactor sites.

An advance notice of rulemaking outlining revisions to present reactor siting criteria was published in the **Federal Register** by the NRC in late July. The proposals show no backing off from the stringent regulations proposed by an August, 1979 report of the NRC's Siting Policy Task Force (NUREG-0625). That document recommended issuing construction permits for reactors intended only for locations where the population density within a 5-mile radius of the plant was 100 persons per square mile or less. From 5 to 10 miles, the limit would be 150 persons per square mile. And for a ring reaching from 10 to 20 miles around the reactor the population density could not exceed 400 per square mile.

Grandfathered, but Vulnerable

Less than half the reactors now in operation or under construction meet these requirements, but these are all grandfathered under the Commission's proposal for a rule. The domestic industry complains, however, that few, if any, prospective sites could meet these criteria.

Moreover, if the rule is adopted, anti-nuclear activists could challenge the existence of present reactors; they could argue that in all but a few cases each and every one was sited according to rules now outdated because of NRC actions in the interest of improved safety, and that therefore these plants are unsafe.

Outside of the U.S., in nations with strong nuclear programs, there is concern that regulatory bodies which traditionally have followed the AEC/NRC lead in safety requirements might adopt the new U.S. criteria, and with similar results. In Japan, for example, where the population density ranks among the highest in the world, nuclear expansion is seen as being halted summarily should the recommended criteria become the rule.

For its part, the NRC, while not oblivious to the problems that might be created by any changes in rules, points out that the essential elements of nuclear power plant siting policy were promulgated by the AEC in 1962, "and have remained essentially unchanged since that time."

Standardizing a Patchwork

Over the years, the NRC has issued additional siting-related pronouncements in the form of siting decisions on specific cases, General Design Criteria, Regulatory Guides, Standard Review Plans, Licensing and Appeal Board decisions, and advice from the Advisory Committee on Reactor Safeguards (ACRS). As a result of this expansion, the NRC admits, "some inconsistencies in staff practice and implementation of the siting regulations have evolved." Through its newly launched rulemaking procedure it hopes to achieve what is seen as badly needed standardization.

The objectives adopted in NUREG-0625, and incorporated in the advance notice of rulemaking, include:

• Strengthening siting as a factor in defense-in-depth by establishing requirements for site approval that are independent of plant design considerations. "The present policy of permitting plant design features to compensate for unfavorable site characteristics has resulted in improved designs but has tended to deemphasize site isolation."

• Taking into consideration risks associated with accidents beyond the design basis by establishing population density and distribution criteria. "Plant design improvements have reduced the probability and consequences of design basis accidents, but there remains the residual risk from accidents not considered in the design basis. Although this risk cannot be completely reduced to zero, it can be significantly reduced by selective siting . . .

• "Siting requirements should be stringent enough to limit the residual risk of reactor operation but not so stringent as to eliminate the nuclear option from large sections of the country. This is because energy generation from any source has its associated risks, with risks from some energy sources being greater than that of the nuclear option."

Forty nine sites in all likelihood could never be expanded in capacity. Thus, about 60 percent of the nation's bank of currently available and approved sites would no longer be usable.

The consensus within the industry is that this last objective is meaningless, because the population density numbers suggested later in the notice would make unacceptable to the NRC most reactor sites seen desirable by the industry.

A review of NUREG-0625 conducted by the Oak Ridge Institute for Energy Analysis concludes that 49 of the existing 84 nuclear power sites in the U.S. could not meet the proposed criteria. In addition, these 49 sites in all likelihood could never be expanded in capacity. Thus, about 60 percent of the nation's bank of currently available and approved sites would no longer be usable.

In a paper on the siting proposal presented in May by Thomas W. Philbin, vice president for environmental and resource planning division of Charles T. Main, Inc., it was also argued; "If the new distance factors are needed to make a site 'acceptable' from the point of view of safety, then the opposite must similarly follow, especially in the minds of a confused public who are being constantly bombarded from all angles by every conceivable type of pro- and anti-nuclear propaganda; i.e., that sites that don't meet the new criteria are 'unacceptable' or 'unsafe.'"

The recommendations, Philbin said, "seem to say that public risk necessitates a specific distance which is far greater than any traditional calculation can justify."

Neither is it realistic, Philbin said, "to allow more or less stringent criteria in differing regions of the country. In both cases, the problem of explaining such differences at hearings make them impractical as well as unjustifiable technically."

Prospect of Regions

Different population density numbers for different regions of the country, although not proposed in the NRC's advance notice of rulemaking, are being considered by the Commission. Shortly after the notice was published, a contract was signed with Sandia Laboratories to perform risk assessments of all nuclear plants using state-of-the-art safety features, including TMI-Lessons Learned modifications.

This information will be incorporated with additional siting factors, such as weather and seismic con-

AT AN AIF INTERNATIONAL BRIEFING ON siting and degraded core issue, Steven Milioti, right, chairman of a Forum subcommittee on emergency preparedness, explains the ramifications of the remote siting criteria proposed by the NRC in late July. The operators of European and Japanese nuclear facilities fear that any NRC standard may be adopted by regulatory bodies in their respective countries, and in some cases eliminate the construction of any new facilities.

siderations, and will permit the NRC to divide the country into regions, each with its own set of population density limits. With this approach, the NRC can meet its objective of avoiding rules that would eliminate the nuclear option "from large regions of the country."

This alternative approach sets no more easily with industry than the hard and fast numbers proposed in NUREG-0625. Steven Milioti, assistant head of the nuclear energy division of American Electric Power Service, is chairman of an Atomic Industrial Forum subcommittee on energency preparedness and siting which is drafting a response to the advance notice. The task is made difficult, says Milioti, because the NRC has no idea at this time of what it means by "region."

"If you define a region large enough—for example, call all of New England one region—then the NRC can come up with the claim that their methodology does not preclude nuclear sites in that region, because there is all of central Maine up there where we can put in reactors. But that's not going to help a Connecticut utility or Boston Edison.

"The NRC will have satisfied its mandate, but in the real world it will have precluded nuclear. So a lot rides on how they define regions if they go that route.

"Were the Commission to adopt a utility's planning region as the regions they use for remote siting, this might be acceptable, but that still leaves the problem of unrealistic population density limits. We understand that the NRC is still intent on coming up with rigid numbers of people per square mile and will say, 'if you're above this limit you cannot site a plant, and if you're below, no questions asked.' Making it different numbers for different regions doesn't solve the problem."

A Zero-Growth Problem

Unexplained by the NRC is its solution to the problem of a site, after it is approved for a reactor, having the area around it experience a population growth that would negate that certification.

The question is posed in the advance notice, but not answered. "What, if any, legislative authority should or could be given to NRC in order to (ensure that) population densities or groupings around nuclear plants remain within acceptable criteria during the operational lifetime of a plant?"

No suggested answers to the question are proposed, but critics of the rule within industry see that situation as a ready-made opportunity for anti-nuclear activists to request a reopening of hearings on a reactor already under construction or in operation.

These concerns are not limited to the U.S. At an AIF International Briefing on Siting and Degraded Core Issues held in late July, several participants from other countries saw a threat to their respective nuclear programs were the NRC population density rule adopted.

Shiro Sasaki, manager of the nuclear power plant operation and maintenance department of Tokyo Electric Power Co., referring to the population density numbers set forth in NUREG-0625, saw that, "depending on the numerical values adopted in the site criteria, the impact could be so far-reaching as to restrict the number of sites that meet the criteria, with the result that the smooth development of nuclear power in Japan could be impeded because of the country's dense population."

Japan: None of the Above

The average population density in Japan is 780 persons per square mile. Said Sasaki: "There is no plant site to meet the new site criteria in the fourteen plant sites which are in operation or under construction . . .

"Establishing the ceiling values for population densities in the criteria creates the difficulty of locating a plant in an area where nuclear power is required. Consequently, it would become necessary to install thermal power plants, with the attendant risks of air pollution to the public, in areas with increasing power demands."

As an alternative approach to increasing the safety factor for populations around reactors, Sasaki supports the recommendations made by U.S. critics of the NRC proposal. These include: "Assurance of public safety by development of engineered safety features, and by isolation of the site; establishment of the design basis accident, and siting is to be handled separately from emergency action.

Said Sasaki, "The risk to the public should be reduced to an acceptable level without having to resort to emergency action. Emergency action is considered to be further reduction of risk. Site selection and evaluation is not contingent upon an emergency plan."

In its advance notice the NRC took pains to down play the possibility that its actions in any way could be viewed as influencing foreign actions. It conceded "that siting criteria in general are matters of national policy as well as national geography and population distribution, and that other nations do not have the same flexibility in siting nuclear facilities as the United States.

"Thus, the Commission wishes to make clear that in emphasizing the use of isolated sites as part of U.S. nuclear siting policy, there is no implication that the siting policies and associated design requirements of other nations result in any less satisfactory protection of the public as judged in the respective national contexts."

Two Commissioners Object

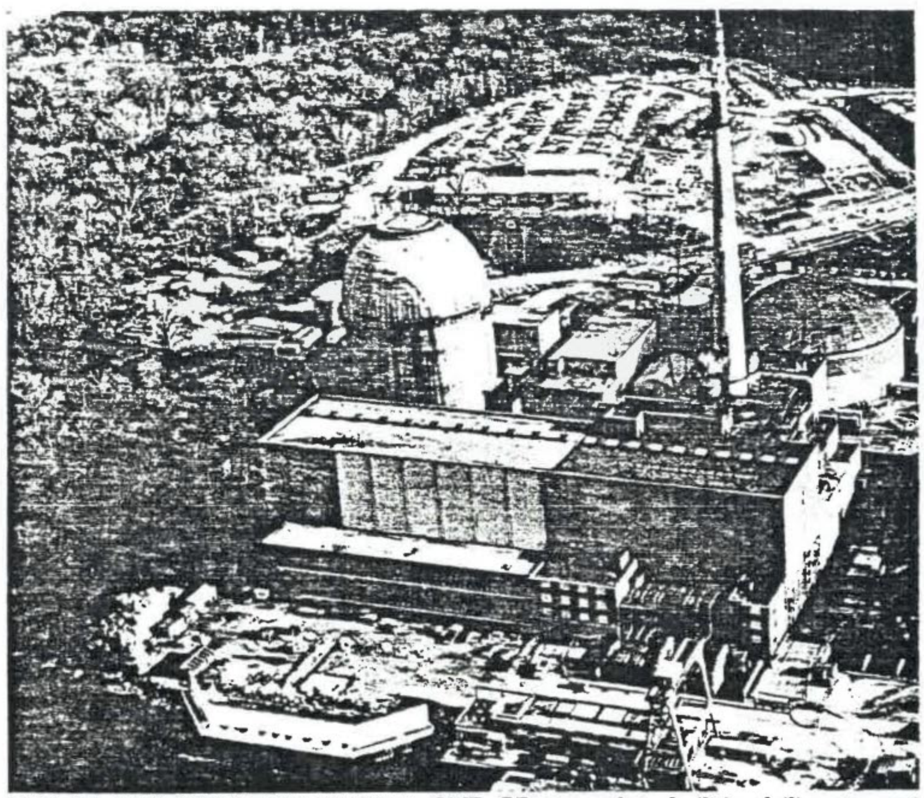
This statement was repudiated by Commissioners Victor Gilinsky and Peter Bradford who commented separately: "We do not think that this reference to the adequacy or inadequacy of siting criteria employed by other countries should be included in this notice.

"Since the NRC has neither jurisdiction over foreign siting criteria, nor any familiarity with foreign sites, these comments are purely gratuitous. Addressing this issue in the context of a rulemaking on domestic siting can only serve to raise questions about the Commission's willingness to temper its protection of the U.S. public so as to accommodate foreign nuclear programs."

Yet other nations, despite the NRC's disclaimer, remain concerned over the potential effects of the possible resulting rule, and will be watching the proceedings closely.

The NRC has asked to receive comments on its advance notice of rulemaking by the end of September. Its present schedule calls for a completed proposed rule to be presented to the full Commission for approval in late February and for publication of the proposed rule in April.

—J.R. Wargo



THE INDIAN POINT COMPLEX IS ONE OF 10 nuclear facilities failing to meet suggested remote siting criteria for population densities within 5, 10 and 20 mile radii. In all, 49 U.S. facilities fail to meet at least one of the three criteria.

Pt, 50

SECRETARIAT RECORD COPY

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

June 19, 1995

MEMORANDUM TO: The Chairman
Commissioner Rogers
Commissioner de Planque
Commissioner Jackson

FROM: James M. Taylor
Executive Director for Operations

SUBJECT: SUMMARY OF PUBLIC COMMENTS RECEIVED ON PROPOSED REVISION OF PARTS 50, 52, AND 100

Attached is a brief history and summary of the public comments received on the second proposed revision of 10 CFR Parts 50, 52, and 100, and a listing of the commentors. As you may recall, the first proposed revision was issued for public comment in October 1992, and subsequently withdrawn. The second proposed revision was issued for public comment in the Federal Register on October 17, 1994 (59 FR 52255). The availability of draft guidance documents for public comment was published on February 28, 1995 (60 FR 10810). The comment period expired May 12, 1995; sixteen commentors responded to these announcements.

In the nonseismic area, several felt that the second proposed revision was an improvement since concerns regarding numerical values of population density and exclusion area distance in the rule had been satisfactorily addressed.

There was general agreement that the use of total effective dose equivalent (TEDE) is warranted. Differences of opinion were expressed on the numerical dose value proposed as an acceptance criterion and on the proposed use of the maximum dose received in any two-hour time period for evaluation purposes.

Most of the comments in the seismic area were supportive of the staff proposal. Many of the comments consisted primarily of editorial and technical suggestions that would clarify the rule or supporting guidance documents. A few of the comments are of a more substantive nature requiring a careful assessment of their implications.

The staff sees no unresolvable points of contention. The staff will be evaluating and resolving these comments, and plans to recommend a final rule to the Commission by the end of October 1995.

Attachment: As stated

cc: SECY ~~OGC OCA OPA ACRS~~

Contact: Leonard Soffer, RES
415-6574
Dr. Andrew J. Murphy, RES
415-6010

6-19-95

SUMMARY OF PUBLIC COMMENTS

10 CFR Parts 50. 52 and 100

Reactor Site Criteria Including Seismic and Earthquake Engineering
Criteria for Nuclear Power Plants
and Proposed Denial of Petition from Free Environment, Inc. et al.

BACKGROUND

The first proposed revision to these regulations was published for public comment on October 20, 1992, (57 FR 47802). The availability of the draft regulatory guides and standard review plan sections that were developed to provide guidance on meeting the proposed regulations was published on November 25, 1992, (57 FR 55601). Because of the substantive nature of the changes to be made in response to public comments the proposed regulations and draft guidance documents were withdrawn and replaced with the second proposed revision of the regulations published for public comment on October 17, 1994, (FR 59 52255). The availability of the draft guidance documents was published on February 28, 1995, (FR 60 10810). The public comment period ended May 12, 1995.

The proposed regulatory action would apply to applicants who apply for a construction permit, operating license, preliminary design approval, final design approval, manufacturing license, early site permit, design certification, or combined license on or after the effective date of the final regulations.

Because the revised criteria presented in the proposed regulation would not be applied to existing plants, the licensing bases for existing nuclear power plants must remain part of the regulations. Therefore, the non-seismic and seismic reactor site criteria for current plants would be retained as Subpart A and Appendix A to 10 CFR Part 100, respectively. The proposed revised reactor site criteria would be added as Subpart B in 10 CFR Part 100 and would apply to site applications received on or after the effective date of the final regulations. Non-seismic site criteria would be added as a new §100.21 to Subpart B in 10 CFR Part 100. The criteria on seismic and geologic siting would be added as a new §100.23 to Subpart B in 10 CFR Part 100.

Criteria not associated with the selection of the site or establishment of the Safe Shutdown Earthquake Ground Motion (SSE) have been placed into 10 CFR Part 50. This action is consistent with the location of other design requirements in 10 CFR Part 50. The dose calculations and the earthquake engineering criteria would be located in 10 CFR Part 50 (§50.34(a) and Appendix S, respectively). Because Appendix S is not self executing, applicable sections of Part 50 (§50.34 and §50.54) are revised to reference Appendix S. The proposed regulation would also make conforming amendments to 10 CFR Part 52. Section 52.17(a)(1) would be amended to reflect changes in 50.34(a)(1) and 10 CFR Part 100.

The following draft regulatory guides and standard review plan sections were developed to provide prospective licensees with the necessary guidance for implementing the proposed regulation:

1. DG-1032, "Identification and Characterization of Seismic Sources and Determination of Shutdown Earthquake Ground Motions." The draft guide provides general guidance and recommendations, describes acceptable procedures and provides a list of references that present acceptable methodologies to identify and characterize capable tectonic sources and seismogenic sources.

2. DG-1033, Third Proposed Revision 2 to Regulatory Guide 1.12, "Nuclear Power Plant Instrumentation for Earthquakes." The draft guide describes seismic instrumentation type and location, operability, characteristics, installation, actuation, and maintenance that are acceptable to the NRC staff.

3. DG-1034, "Pre-Earthquake Planning and Immediate Nuclear Power Plant Operator Post-Earthquake Actions." The draft guide provides guidelines that are acceptable to the NRC staff for a timely evaluation of the recorded seismic instrumentation data and to determine whether or not plant shutdown is required.

4. DG-1035, "Restart of a Nuclear Power Plant Shut Down by a Seismic Event." The draft guide provides guidelines that are acceptable to the NRC staff for performing inspections and tests of nuclear power plant equipment and structures prior to restart of a plant that has been shut down because of a seismic event.

5. Draft Standard Review Plan Section 2.5.1, Proposed Revision 3, "Basic Geologic and Seismic Information." The draft describes procedures to assess the adequacy of the geologic and seismic information cited in support of the applicant's conclusions concerning the suitability of the plant site.

6. Draft Standard Review Plan Section 2.5.2, Second Proposed Revision 3 "Vibratory Ground Motion." The draft describes procedures to assess the ground motion potential of seismic sources at the site and to assess the adequacy of the SSE.

7. Draft Standard Review Plan Section 2.5.3, Proposed Revision 3, "Surface Faulting." The draft describes procedures to assess the adequacy of the applicant's submittal related to the existence of a potential for surface faulting affecting the site.

8. DG-4003, Second Proposed Revision 2 to Regulatory Guide 4.7, "General Site Suitability Criteria for Nuclear Power Plants." This guide discusses the major site characteristics related to public health and safety and environmental issues that the NRC staff considers in determining the suitability of sites.

SUMMARY OF COMMENTS ON REACTOR SITING CRITERIA (NONSEISMIC)

Eight organizations or individuals commented on the nonseismic aspects of the proposed revisions. The first proposed revision issued for comment in October 1992 elicited strong comments in regard to proposed numerical values of population density and a minimum distance to the exclusion area boundary (EAB) in the rule. This second proposed revision would delete these from the rule by providing guidance on population density in a Regulatory Guide and determining the distance to the EAB by the use of source term and dose calculations. Several commentors representing the nuclear industry and international nuclear organizations stated that this was a significant improvement over the first proposed revision, while the only public interest group commented that the NRC had retreated from decoupling siting and design in response to the comments of foreign entities.

Most comments on the second proposed revision centered on the use of total effective dose equivalent (TEDE), the proposed single numerical dose acceptance criterion of 25 rem TEDE, the evaluation of the maximum dose in any two-hour period, and the question of whether an organ capping dose should be adopted. Virtually all agreed that the concept of TEDE was appropriate and should be used. However, there were differing views on the proposed numerical dose of 25 rem and the proposed use of the maximum two-hour period to evaluate the dose. Virtually all industry commentors felt that the proposed numerical value was too low and that a "sliding" two-hour window for dose evaluation was confusing and inappropriate. All industry commentors opposed the use of an organ capping dose. The only public interest group that commented did not object to the use of TEDE, and believed the proposed dose value of 25 rem to be appropriate, but favored an organ capping dose. A summary of each commentors remarks follows.

Adams Atomic Engines, Inc.

There is no need to have a rule based on a traditional requirement to keep nuclear power plants far from population centers. Remote siting criteria are no longer necessary. The proposed rule has the potential for negative economic, environmental, and safety impacts on the general public, reactor suppliers and power plant operators.

The source term should be based on a maximum credible accident instead of an assumed "substantial meltdown of the core".

Ohio Citizens for Responsible Energy, Inc. (OCRE)

The proposed rule is unacceptable with respect to the nonseismic criteria. The NRC has retreated from decoupling siting and design as proposed in October 1992, in response to comments from foreign entities.

OCRE believes that the footnote in the proposed section 100.21(h) about considering economic factors is improper under Atomic Energy Act, since NRC may not consider costs to licensees.

OCRE has no objection to the use of TEDE; this is necessary if the new source term is to be used. The appropriate acceptance value is 25 rem. The NRC should also adopt an organ "capping" dose. No more than 35% of the total dose should be from a single organ.

Northeast Utilities Service System

Adopting dose criteria in terms of TEDE is consistent with recent guidance (ICRP, EPA). TEDE captures the overall potential health consequences, and is the most practical approach for limiting the combined effect to all organs. 25 rem is appropriate and consistent with the value established in other guidance documents, such as EPA 400, as an acceptable exposure to an individual.

An organ "capping" dose is not necessary since design basis accidents do not involve only iodine.

The requirement to determine the maximum two-hour period (for dose calculation) is not practical, nor necessary. It unduly complicates the radiological analysis. The concept also questions the resulting conclusion. If an individual received less than 25 rem from an exposure from 30 minutes to 2 hours and 30 minutes, what about the dose received before the 30 minute period?

ABB Combustion Engineering Nuclear Systems

Expresses concern that a site approved under present Part 100 for a currently operating reactor might not be approved for an advanced light water reactor (ALWR) under the proposed Part 100. This presents a quandary since ALWR has improved safety features.

ABB-CE fully supports the use of TEDE. The proposed dose limit was first estimated at 27 rem. NRC staff adjusted this downward to 25 rem without explanation. The value of 27 rem is more appropriate; however, the development of a more technically justifiable criterion should be pursued.

It is not clear that cancer risk is the best parameter for maintaining same level of protection. Offsite dose limit does not represent an acceptable dose to any member of the public, but is a "figure of merit". The activity corresponding to the current 300 rem thyroid and 25 rem whole body should be calculated for conservative weather conditions. ABB also strongly believes that the dose acceptance criterion should also reflect consideration of any contribution from the additional nuclides identified in NUREG-1465.

Advanced Light Water Reactor (ALWR) Program

The ALWR Program supports the use of TEDE, but not a dose acceptance criterion of 25 rem. Based on organ weighting factors given in ICRP 26, 25 rem whole body and 300 rem thyroid are equivalent to a value of 34 rem TEDE. Based on organ weighting factors given in ICRP 60 (using a revised thyroid weighting factor), the current dose criteria are equivalent to 40 rem TEDE.

There is no need for an organ capping dose, since iodine is unlikely to be present by itself.

ALWR suggests as an alternate criterion that the dose at the exclusion area boundary (EAB) should not exceed 40 rem TEDE over a 24 hour period. Also proposes significant changes in the way that meteorology dispersion factors (X/Q) are calculated, since the present approach in Reg. Guide 1.145 is overly conservative.

If 2 hour dose calculation is retained, it should begin with the start of the accident which should be defined as no later than the start of the gap release to the containment. While this does not tie the dose calculation to the declaration of a General Emergency, it reflects that reality far better than a sliding 2 hour window.

Nuclear Energy Institute (NEI)

NRC staff is to be congratulated for carefully considering and responding to complex public comments on the first proposed revision. Many troubling aspects of the first proposed revision have been addressed in a forthright and appropriate manner.

NEI supports the use of TEDE. This will support a uniform and consistent implementation of realistic source terms. A value of 25 rem is more restrictive than the current dose criteria, and does not represent the total stochastic risk, because the value of 300 rem thyroid is about 9 rem TEDE. NRC should determine the appropriate numerical value utilizing the total stochastic risk implied by the current criteria, and this should be incorporated without additional conservatism or adjustment.

An organ capping dose limit is not practical nor necessary.

There is little justification for changing the 0 to 2 hour dose calculation, at least partly because other aspects of the calculation (e.g., meteorology) are not yet clear as to how they would be calculated.

NEI supports NRC's proposed approach with regard to population density criteria, as stated in draft Regulatory Guide DG-4004. NEI supports the concept of environmental justice, but expresses concerns regarding subjective phrases and potential implementation. Recommends that the environmental justice provision be deleted from this revision of the Guide until more detailed guidance becomes available.

Morgan, Lewis and Bockius

The 1994 proposed rule is a major improvement over the previous version.

Morgan, Lewis and Bockius expresses concerns in regard to 2 changes. The proposed dose acceptance criterion of 25 rem TEDE could make NRCs accident dose limits significantly more restrictive, without any showing that these are necessary to protect public health and safety.

The proposed change from an immediate 2 hour period to a moving 2 hour period will impose another unidentified penalty, depending upon the design. Also, the change is contrary to common sense, since it requires that during an accident a member of the public will move toward the plant rather than away from it.

Westinghouse Electric Corporation

The proposed "sliding dose window" is not linked to any specific occurrence, and ignores any dose accumulated during the time between accident initiation and the two hour interval of highest dose. A more reasonable approach would be to replace it with a time interval of two hours starting with the onset of core damage plus the time interval between accident initiation and the onset of core damage. Westinghouse also proposes consideration of an additional dose criterion that the 24 hour dose at the exclusion area boundary (EAB) should not exceed twice the acceptable 2 hour dose at the same location.

Endorses the use of TEDE, but believes that the risk associated with the current dose limits would support a significantly higher numerical dose value than the value of 25 rem proposed. There is no need for an organ "capping" dose, which would result in an unnecessary complication without reducing risk to the public.

SUMMARY OF COMMENTS ON SEISMIC AND EARTHQUAKE ENGINEERING CRITERIA

A total of eleven individuals or organizations commented on the proposed revisions. A general assessment of the comments is that most are supportive of the staff positions. Many of the commentors have provided editorial and technical suggestions that would clarify the rulemaking. A few commentors provided more substantive comments requiring a careful assessment of their implications. The following is a summary of each commentor's input with focus principally on their recommendations.

American Society of Civil Engineers (Washington Office)

The seismic design and engineering criteria of ASCE Standard 4, "Seismic Analysis of Safety-Related Nuclear Structures and Commentary on Standard for Seismic Analysis of Safety-Related Nuclear Structures," should be incorporated by reference into the regulation.

G.C. Slagis Associates

Comments are limited to pressure-retaining components to the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Section III rules. Questions the soundness of only the Safe Shutdown Earthquake Ground Motion (SSE) being used for design, that is, the elimination of Operating Basis Earthquake Ground Motion (OBE) response analyses. Also, provided technical comments on supplementary NRC staff positions on fatigue analysis (positions established in certification review of ALWRs) and post-earthquake inspections (DG-1034, DG-1035).

Wais and Associates

Commends the NRC staff for adopting the probabilistic seismic hazard approach versus the deterministic approach for the Central and Eastern United States.

Site investigations are performed at four levels with the amount of detail based on distance from the site. Recommends reducing the outer area of geological and seismic investigations (DG-1032) and not restricting the updating of the LLNL and EPRI probabilistic seismic hazard databases to only situations that lead to higher hazard estimates (DG-1032). Questions the logic used to define the reference probability for the SSE exceedance level (Appendix B to DG-1032). Also questions the need for seismic instrumentation (DG-1033), and the need for plant shutdown if the OBE is exceeded and no damage is apparent (DG-1034).

ABB-CE

Agrees with the NRC staff's proposal to not require explicit design analysis of the OBE if its peak acceleration is less than one-third of the SSE.

Department of Energy (Office of Civilian Radioactive Waste Management)

Requests an explicit statement whether or not the proposed regulations apply to the Mined Geologic Disposal System (MGDS) and a Monitored Retrievable Storage (MRS) facility. Site investigations are performed at four levels with the amount of detail based on distance from the site. Recommends that the stated outer area of investigations should be reduced and that the applicant should justify its rationale for the area of investigations considered (DG-1032, SRP Sections 2.5.1 and 2.5.2).

Nuclear Energy Institute

Congratulates the NRC staff for carefully considering and responding to the voluminous and complex comments that were provided on the earlier proposed rulemaking package and considers that the seismic portion of the proposed rulemaking package is nearing maturity and with the inclusion of industry's comments, has the potential to satisfy the objectives of predictable licensing and stable regulations.

Supports the regulation format, that is, prescriptive guidance is located in regulatory guides or standard review plan sections not the regulation.

Supports the removal of the requirement from the first proposed rulemaking that both deterministic and probabilistic evaluations must be conducted to determine site suitability and seismic design requirements for the site. However, does not agree with the NRC staff's deterministic check of the seismic sources and parameters used in the LLNL and EPRI probabilistic seismic hazard analyses (DG-1032). Also, does not support the NRC staff's deterministic check of the applicants submittal (SRP Section 2.5.2).

The regulation and guidance documents should state that if an ALWR is to be sited at an existing nuclear power plant site, only confirmatory investigations of foundation conditions are required (Regulatory Guide 1.132). Also, state that for existing sites east of approximately 105° west longitude a 0.3g standardized design level is acceptable.

For nuclear power plants founded on rock sites the licensee should have the option to use the containment basemat data (instead of free-field data) to determine OBE exceedance (DG-1034).

Provided over 60 specific technical or editorial comments on the seismic portion of the rulemaking (regulation, regulatory guides and standard review plan sections).

Morgan, Lewis & Bockius

Concerned with the emphasis on the probabilistic analysis to establish the SSE. Although Section 100.23 states that a suitable sensitivity analysis can be used to address uncertainties in the SSE, DG-1032 contains no discussion for addressing uncertainties in the SSE except for performing a probabilistic seismic hazard analysis (PSHA). Also, there is no clear statement in DG-1032 that if a PSHA is performed no further analysis is necessary or if a suitable sensitivity analysis is performed a PSHA is not necessary.

Yankee Atomic Electric Company

At existing Eastern United States sites (rock or soil) or at rock Eastern United States sites not located in areas of high seismicity (for example, Charleston, South Carolina, New Madrid, Missouri, Attica, New York) a 0.3g standardized ALWR design is acceptable and only evaluations of foundation conditions at the site are required (Regulatory Guide 1.132) but not geologic/geophysical seismological investigations. For other sites a DG-1032 review is required.

Proposes an alternative to DG-1032 that incorporates soil amplification into the probabilistic analysis, does not allow scaling of the SRP Section 2.5.2 site specific spectra to define the SSE, but allows the scaling of broad-banded spectra to define the SSE.

Kinemetrics, Inc.

In general, agrees with Draft Regulatory Guide DG-1033. However, cannot comply with the battery capacity recommendations in the draft guide. Also, recommends that Regulatory Position 4.3 of an earlier draft regulatory guide (DG-1016) addressing the interconnection of instrumentation for common starting and common timing be reinstated in the final guide.

TU Electric

The recommendation for fatigue analysis in Regulatory Position 1.2 of DG-1035 should be limited to ASME Code Class 1 components and systems. Also, clarify Regulatory Position 1.3 in DG-1035, the analysis recommendation for non-safety related systems and components.

Westinghouse Electric Corporation

Supports NRC staff decision to move guidance material from the rule to regulatory guides. Supports NRC staff decision to eliminate the "dual" deterministic and probabilistic analyses from the proposed rule. Concerned that retaining deterministic evaluations in SRP Section 2.5.2 will lead to confusion as to whether future licensees will also need to perform a deterministic analysis even though such an analysis is only recommended for NRC staff to perform as a "sanity" check. Shares NEI's concern with respect to the type of analyses needed to construct a new plant on an existing approved site, using the proposed rule and associated regulatory guides.

List of Commentors on Proposed Revision of Parts 50, 52, and 100*

Number	Commentor	Nonseismic	Seismic	Both
1	Adams Atomic Engines, Inc.	X		
2	American Society of Civil Engineers (ASCE)		X	
3	Ohio Citizens for Responsible Energy (OCRE)	X		
4	G. C. Slagis Associates		X	
5	Northeast Utilities	X		
6	Wais and Associates		X	
7	Wais and Associates		X	
8	ABB Combustion Engineering Nuclear Systems			X
9	Advanced Light Water Reactor Program (ALWR)	X		
10	U. S. Department of Energy		X	
11	Nuclear Energy Institute (NEI)			X
11A	Nuclear Energy Institute -- Supplementary	X		
12	Morgan. Lewis and Bockius			X
13	Yankee Atomic Electric Company		X	
14	Kinemetrics, Inc.		X	
15	TU Electric		X	
16	Westinghouse Electric Corporation			X

* Does not include requests for extension of the comment period.

O'M 7ACRS

SECY 96-0414

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

May 24, 1996

Dr. T.S. Kress, Chairman
Advisory Committee on Reactor Safeguards
U.S. Nuclear Regulatory Commission
Washington, DC 20555

Dear Dr. Kress:

SUBJECT: ACRS LETTER DATED APRIL 22, 1996 - PROPOSED REVISIONS TO 10 CFR PARTS 50 AND 100 AND PROPOSED REGULATORY GUIDES RELATING TO REACTOR SITE CRITERIA

I am responding to your letter of April 22, 1996, regarding the proposed revisions to 10 CFR Parts 50 and 100 and proposed regulatory guides relating to reactor site criteria. In your letter, you recommended that the proposed final rule and the associated Regulatory Guides and Standard Review Plan sections be issued, both with respect to the seismic and non-seismic aspects. With regard to the non-seismic aspects, you also agreed with the position taken in the proposed rule relative to the time period for the dose evaluation. You did, however, recommend that careful definitions of the TEDE limits be included in the final rule that are mindful of organ dose weighting factors found in 10 CFR Part 20. You further noted that consistency within the body of NRC regulations is "most desirable".

As discussed in the draft Commission paper and proposed Statement of Considerations, the staff is recommending 25 rem TEDE as the dose criterion. The staff came to that conclusion from a number of perspectives. The current Part 100 has, as criteria, values of 25 rem to the whole body and 300 rem thyroid for a hypothetical individual located at the exclusion area boundary. As stated in the Statement of Considerations for the proposed rule, this converts to 27 rem TEDE when the conversion is made on the basis of maintaining equivalent risk between the current and proposed criteria using the currently accepted risk coefficients for latent fatalities. In addition, in terms of occupational dose, the revised Part 20 permits a once-in-a-lifetime planned special dose of 25 rem TEDE. Also, EPA guidance sets a permissible dose limit of 25 rem TEDE for emergency radiation workers. The above information led the staff to conclude that selecting 25 rem TEDE as the dose criterion was appropriate.

A number of commenters on the proposed rule noted that the use of organ weighting factors from 10 CFR Part 20 and the current limits of 25 rem whole body and 300 rem to the thyroid would yield a value of 34 rem TEDE. While the use of the 10 CFR Part 20 organ weighting factors does lead to a value of 34 rem TEDE, this is because the organ weighting factors in 10 CFR Part 20 include other effects (e.g., genetic) in addition to latent fatalities. Therefore, as stated above, the appropriate dose based upon latent fatality risk conversion is 27 rem TEDE. In addition, the argument that 25 rem TEDE represents a tightening of the dose criterion is not true in practice.

5/24/96

As stated in the proposed Statement of Considerations for the subject rule, a review of the dose analyses for operating plants has shown that the thyroid dose limit of 300 rem has generally been the limiting dose criterion in licensing reviews, and that all operating plants would be able to meet a dose criterion of 25 rem TEDE using updated source term insights. Since the proposed 25 rem TEDE criterion would allow higher doses to the thyroid, provided the overall dose met the 25 rem TEDE limit, the staff does not consider the 25 rem TEDE value a tightening of the criterion. To maintain consistency within the body of regulations the staff proposes to use the organ weighting factors currently in 10 CFR Part 20 for converting calculated organ doses to TEDE for comparison to the 25 rem TEDE criterion, even though the Part 20 organ weighting factors are based upon other considerations in addition to latent fatalities. As discussed above, from a practical stand point, this does not represent a tightening of the dose criterion. A future update to Regulatory Guides 1.3 and 1.4 will provide guidance on the dose calculation methods to be used.

Sincerely,

James M. Taylor
Executive Director
for Operations

cc: Chairman Jackson
Commissioner Rogers
Commissioner Dicus
SECY