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ATOMIC ENERGY COMMISSION UNDTD

REACTOR SITE CRITERIA

Report to the General Manager by the
Director, Division of Licensing and Regulation

THE PROBLEM

1. To consider criteria proposed for use in the approval of sites for licensed power and test reactors, to explain the basis upon which the criteria were established, and to provide an understanding of the relative safety to the public that will result from application of the criteria in the site selection process.

SUMMARY

2. An applicant for a license to construct a power or test reactor is required by AEC regulations (10 CFR Part 50) to submit in support of his application a hazards summary report that includes details pertinent to the site proposed for the reactor. The current regulations do not indicate how the site data supplied by applicants will be evaluated by the AEC, or the specific criteria which will guide the AEC's consideration of proposed site suitability.

3. For reactors that have already been proposed, site approval or disapproval has been given after review and evaluation of the reactor design and the proposed location by the staff of the Division of Licensing and Regulation and the ACRS. Judgment has been based primarily upon the evaluation of the consequences of potential accidents, including an accident representing an upper limit of hazard that could credibly occur. This evaluation process has also included analysis of the plant design and particularly the safeguards either inherently

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part of the reactor or engineered into the plant complex for safety reasons.

4. The hazards reports as presented by the various applicants have shown a wide variation in estimating the magnitude of the maximum credible accident and in the dose calculational methods and, consequently, in the calculated exposure doses that might result to the offsite public in case of an accident. This situation is due partly to the differences in reactor plant design but even more to the different engineering judgments that can be made in analyzing possible consequences of accidents. AEC and ACRS review has emphasized evaluation of the safety factors that have been included in the plant design and evaluation of the conservatism represented in the analytical procedures as well as the numerical values derived. This subjective manner of arriving at judgment on site suitability has led to requests to have the AEC make more definitive the basis upon which the data are evaluated and to make more specific the safety criteria which govern the AEC's consideration of site suitability.

5. An attempt was made in May 1959 to establish a more objective approach to reactor site selection and evaluation by publishing proposed site criteria in the Federal Register. The reactions of the industry were widespread; most of those who commented were opposed to the proposed regulation but the reasons for the opposition were quite heterogeneous. The criteria proposed in 1959 and excerpts of written comments on them received by the AEC are included in information paper AEC-R 2/20. It would appear from these comments that the industry, while pressing for criteria that would define the conditions of acceptability for proposed reactor sites, want such information in the form of guides but not in the form of a regulation.

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6. The JCAE has shown continued interest over the past several years in AEC efforts toward formulating more definitive site criteria. During the hearings before the Subcommittee on Research and Development and the Special Subcommittee on Radiation of the JCAE on April 27, 1960, the criteria published by the AEC in the Federal Register in May 1959 were discussed with particular reference to the role of those criteria in the evaluation of a proposed reactor site at Jamestown, New York. Regarding the shortcomings of these earlier criteria, Chairman McCone expressed the view that the problem of site criteria was one that must be settled in order that builders of nuclear power plants might proceed with more assurance and that clarification of AEC site requirements appeared possible in the very near future. At that same hearing, Dr. C. R. McCullough, as a representative of the ACRS, stated that the ACRS believed the time had come to put site criteria in writing.

7. In December 1959, the General Manager established a special working group, in which experts from industrial organizations were included, to examine the question of what the Commission could and should do in the way of establishing standards and criteria in the field of nuclear safety. (This fact was reported by Commissioner Graham to the JCAE during the 202 hearings in February 1960.) In a report to the General Manager dated September 29, 1960, (AEC-R 2/21) this Ad Hoc Committee recommended that the Commission "establish rules, involving of necessity some degree of arbitrariness, by which sites that would be considered acceptable for locations of reactors could be selected."

8. Proposed criteria (Appendix "D") have been prepared that describe the bases upon which the suitability of proposed reactor

sites can be judged. As a beginning point, the criteria define three bench marks, stated in terms of areas and distances, for evaluation of proposed sites for a reactor of any given power level. These are (1) an exclusion area over which the licensee controls the access; (2) a zone surrounding the exclusion area in which the density of population is sufficiently low to permit evacuation in case of a castastrophic accident; and (3) a distance to the nearest population center in which more than 25,000 people reside. These areas and distances are determined upon the following assumptions: (1) in establishing the exclusion and evacuation distances, the amount of radioactivity released to the environment will not exceed that expected from the accident considered to be "the maximum credible accident"; (2) within the exclusion area the operator will have full control and may take whatever steps are necessary to protect any people who may be therein; (3) the radiation dose to persons within the evacuation area may be limited by evacuation or other counter-measures sufficiently to prevent immediate or early manifestation of radiation injury; and (4) the population center distance is calculated on the assumption that persons in nearby centers of population would not be lethally exposed in the event of an accident similar to the maximum credible accident but in which no containment or retention whatever of the released fission products were accomplished by the reactor building. Iodine doses such as those specified (in later sections) on the basis of these premises, if actually received by people, do not preclude the possibility of the production of a number of cases of leukemia or cancer in later years. However, it is believed that in view of the small probability of occurrence of accidents comparable to the "maximum credible accident", the hazard from such effects as well as from genetic effects is reasonably small. The criteria then provide for adjustment of these bench mark distances in each case in accordance with the unique features and circumstances of that individual reactor project. The proposed rule makes it clear that the bench mark distances are only a beginning point for preliminary guidance and have to be considered along with other equally important factors.

9. Draft criteria along the lines of those proposed in Appendix "D" were forwarded to the ACRS for review and comments. A copy of that draft is contained in AEC-R 2/22. By letter to the Chairman, AEC, dated September 26, 1960, (attached as Appendix "C-1") the ACRS commented on the proposed criteria by stating that "while the Committee believes that the present document could be developed into a useful contribution to nuclear safety studies -- we cannot recommend that it be given the status of a Commission regulation." A similar recommendation is made in a letter of October 22, 1960, from the ACRS to Chairman McCone (Appendix "C-2"). This letter, which also contains other material relevant to site criteria, is discussed further in Appendix "A".

10. There is no disagreement between the ACRS and the staff on the methods and the approach to site evaluation. An effort has been made in the present revised draft of the regulation to take account of all the technical comments on the ACRS. The values stated in the ACRS letter have been used in the regulation except that we know of no practical way to deal with the concept of total population (man rem) dose limitations, but we do believe that the objective of the ACRS on this point is substantially achieved by the criteria proposed. The staff does not, however, agree with the ACRS recommendation that no regulation on the subject of site criteria should be published. The proposed regulation (Appendix "D") contains the same general approach to site criteria as the draft submitted to the ACRS. However, it has been modified to use the numbers recommended by the ACRS and to allow more flexibility in its use.

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11. The proposed criteria represent a simplification of the complex technical problem that site selection presents and do not eliminate a large element of subjective judgment by the evaluators. Nonetheless, the criteria would give the industry, local health and safety authorities and the public a much clearer understanding of what the AEC does with the site information submitted for review, and the elements considered when site suitability is to be judged. The staff believes that the criteria reflect a conservative approach to the problem of siting of reactors with respect to potential hazards to surrounding populace. Should the Commission so desire, the criteria could be revised to reflect either more or less conservatism with respect to degree of isolation to be required in future reactor projects.

STAFF JUDGMENTS

12. The Division of Biology and Medicine, the Division of Reactor Development, the Office of General Counsel, and the Office of Health and Safety concur in the recommendation of this paper.

RECOMMENDATION

13. The General Manager recommends that the Atomic Energy Commission:

a. Approve publication in the Federal Register, for comment, of the proposed Part 51 "Criteria for the approval of Sites for Power and Testing Reactors", attached as Appendix "D";

b. Note that a copy of the proposed regulation will be sent to the Joint Committee;

c. Note that an appropriate news release will be issued;

d. Consider the advisability of Commission discussion with the ACRS and subsequent review by the Commission before any of the foregoing actions are completed;

e. Note that this paper is unclassified.

LIST OF ENCLOSURES

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APPENDIX "A"

BACKGROUND

Introduction

1. The Atomic Energy Act did not lay down any specific criteria to be followed in the issuance of reactor licenses but left to the AEC the definition of such standards as it felt necessary to govern the design, location, and operation of nuclear facilities "in order to protect health and minimize danger to life and property." The regulations issued to date by the AEC pertinent to reactor siting (10 CFR 50) deal principally with the information that must be submitted in support of license applications. This information is required to be submitted as a part of a "hazards summary report" and includes the following:

a. A description of the processes to be performed in the reactor and the nature and quantity of radioactive effluents expected to be produced.

b. A description of the facility in sufficient detail to allow evaluation of the adequacy of measures to minimize danger to persons both on-site and off-site.

c. A description of the site and the surrounding area, including pertinent meteorological, hydrological, geological and seismological data necessary for evaluating measures proposed for protecting the public from radioactive hazards.

2. Current regulations do not indicate, however, how the data supplied will be evaluated by the AEC, or the safety criteria which govern the AEC's consideration of proposed site suitability. Thus a prospective reactor plant builder is provided with little in the way of definitive guidance during the initial selection of a reactor site nor can he plan with any assurance during the period his proposed site is under review by the AEC. Local safety authorities and the public near such reactor sites likewise have little to base judgment on as to how

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their interests are being protected other than a general awareness that within the AEC such projects are being reviewed with welfare of the public in mind.

3. One of the consequences of Commission silence regarding reactor site criteria policies is the possibility of development of divergent approaches and philosophies by various segments of the AEC involved in siting problems.

4. It is generally recognized that uncontrolled release to the atmosphere of the radioactive contents of a reactor system located in a densely populated area would result in public disaster. This awareness has led to the provision in the past of a considerable isolation area surrounding reactor installations. This was done on the theory that if enough distance was provided between a reactor and the perimeter of the controlled area, little or no jeopardy to the public would be involved.

5. The earlier concept of remoteness for reactor locations has undergone modification to the extent that plants with less isolation coupled with containment vessels have been judged adequate to protect the public health and safety. Although this change in concept is in the direction of bringing reactor plants closer to the demand centers, the nuclear power industry for economic reasons still presses for a further reduction in the conservatism inherent in such a concept.

6. Any further reduction in the concepts of isolation and containment for reactors will be largely dependent upon the ability to assess with more certainty the circumstances and conditions under which loss of control of radioactive inventory might arise and the possible consequences of such an accident. The process of hazard analysis and site selection at this stage of technology is not a precise science, for the many variables

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involved are not precisely known nor has experience been sufficient to provide exact knowledge about the degree of conservatism that exists in past assumptions and guiding design criteria.

Present Practices in Site Evaluation

7. Judgment of suitability of a reactor site for a nuclear plant is a complex task. In addition to normal factors considered for any industrial complex such as nearby land use, water supply, soil and underlying rock characteristics, and site accessibility, are engineering features dictated by reactor hazards, including the hazards of radioactivity which vary with the type and size of plant to be built and the manner in which the potential radioactive effluents could be carried to the public.

8. A somewhat greater susceptibility to nuclear accidents might be attributed to test reactors versus power reactors because of the different utilization of the nuclear energy generated. However, the "upper limit of hazard" represented by the maximum credible accident is no greater for a test reactor than a power reactor of the same size, and is frequently less since the energy that is stored within the coolant system of the test reactor is less. However, the similiarities between power and test reactor are considered sufficient to justify consideration of their hazards by common standards.

9. Proposed sites for power and test reactors are evaluated by both the staff of the Division of Licensing and Regulation and the ACRS. Information supplied by the applicant is reviewed for answers to such questions as the following:

a. What is the size of the site and the location of the reactor on the property? This information fixes the exclusion radius for the reactor with respect to the nearest uncontrolled land.

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b. What is the industrial and population distribution in the surrounding areas? This information is important in assessing the consequences of inadvertent release of radioactivity. The size of the required exclusion area will be affected by many factors including among other things reactor power level, design features and containment and site characteristics.

c. What are the relevant features of hydrology, including location and number of nearby sources of drinking water or bathing facilities? This factor is important in evaluating the liquid waste disposal facilities proposed by the applicant. For example, the hydrology of the ground waters is important in assessing the effect travel time may have on the contaminants which might reach them to the points of nearest usage. Site drainage and surface water is important in determining the vulnerability of surface water sources to radioactive contamination. The characteristics and usage of the water sources often determine the safety precautions that must be observed at the facility in effluent control and management. decay during travel

d. What are the significant meteorological factors? The persistence of inversions, the prevailing wind directions and velocities, and the rainfall become significant parameters in considering effects of airborne radioactivity. Capabilities of the atmosphere to diffuse and disperse an airborne release are considered in assessing the vulnerability to risk of the areas surrounding the site. Thus, a high probability of good diffusion conditions and a wind direction pattern away from vulnerable areas during periods of slow diffusion would enhance the suitability of a site. On the other hand, if a site were in a region noted for hurricanes or tornadoes, it would be expected that the design of the facility include safeguards which would prevent significant radioactivity releases should one of those events occur.

e. What has been the history of seismological disturbances in the area? Certain areas in the U. S. are known to have active faulted sub-surface structure and the requirements for buildings in such an area need added attention to possible consequences of ground tremors and shocks.

f. What is the soil structure for the site? This factor is important not only to design of the structural aspects of the facility but also to safety aspects relating to liquid waste storage and disposal. Highly permeable soils for example could lead to contamination of sub-surface aquifers from leaking storage containers. Impermeable soils on the other hand might lead to quick and uncontrolled runoff of liquid spills into nearby streams.

10. All the factors described are interrelated and dictate in varying degrees the engineered protective safeguards required for an individual facility. Therefore, site evaluation also includes consideration of the general features of the reactor

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plant including power level, general plan of utilization and the safeguards planned to preclude or minimize inadvertent release of radioactive effluents.

11. An analytical test of the safeguards provided by site location and plant design is made through evaluation of a postulated accident, having consequences not expected to be exceeded by any other accident arising out of any other credible circumstances. Analysis is made of the consequences in terms of possible radiation exposure both to personnel at the facility and to the inhabitants of the surrounding public area. The conservatism of the assumptions made in arriving at the results and the acceptability of characteristics attributed to the safeguards provided are considered in assessing the numerical values derived. The judgment made is thus highly subjective. The many variables involved are not precisely known nor has experience been sufficient to provide exact knowledge about the degree of conservatism that exists in past design assumptions and guiding criteria.

History of the Problem

12. Attempts to become more objective through the use of definitive criteria have been complicated by a variety of situations including the following:

a. The industry, while pressing for criteria that would define the conditions of acceptability of proposed reactor sites, does not want such criteria in the form of regulations but rather in the form of "guides."

b. The end objective in controlling reactor site location is to provide reasonable assurance that the public will not be subjected to undue hazards from operation of the facility. Any meaningful evaluation of the hazard associated with a particular accident must take into account the probability that the accident will occur, the resulting severity of exposures of individual persons to radiation, and numbers of persons at risk. While one cannot make quantitative and detailed evaluation of these factors, the present approach attempts to give to each the greatest consideration presently practicable. The probability of severe accidents is considered to be limited by technical reviews of reactor design and specifications, by conditions of license, and by inspection. Limitations of numbers of persons at risk are provided by exclusion, evacuation, and population center boundaries. Limits imposed on corresponding radiation doses are necessarily arbitrary since the related factors of probability of accident and numbers of persons cannot be closely defined. For the purposes of these criteria we have selected as limits doses which would not result in early manifestations of injury in case of the maximum credible accident and which are believed to involve a reasonably small probability that any individual receiving such a dose would suffer a serious consequence (such as leukemia or cancer) in later years.

The dose limits specified are 25 rem to the whole body and 300 rem to the adult thyroid. The degree of hazard associated with a dose of 25 rems to the whole body or to a major portion of the body has been qualitatively characterized in a statement by the NCRP that an accidental or emergency dose received only once in the lifetime of a person need not be included in the determination of the exposure status of the person exposed. There is no equivalent recommendation for evaluation of accidental dose to the thyroid. On the basis of staff discussions, 300 r to the adult* thyroid has been used in these criteria.

c. The analysis techniques applied to evaluation of hazards of reactor plant catastrophes cannot be considered to be precise. Experimental verification of parameters used is lacking and will probably remain so for years to come. As a consequence, both designers and evaluators have introduced conservative safety factors. There occurs, nevertheless, considerable variation in calculated results because of the different factors used. No one set of assumptions can be established as exact and appropriate to all situations. Appendix "B" presents further information on the factors involved and the effects on calculations of potential radiation hazards at the site boundaries and selected points beyond.

13. Notwithstanding these deterrents to the formulation of definitive site criteria the AEC has been attempting to establish a more objective approach to site evaluation. For example, the AEC issued for public comment and published in the Federal Register on May 23, 1959, a notice of proposed rule making that set forth general criteria for evaluation of sites for power and test reactors. That notice resulted in widespread reactions from

*If only adults were involved, the thyroid dose could be much higher. It is currently believed that (1) exposures resulting in a dose of this magnitude to the adult thyroid are likely to result in doses some two or three times as high in very small children; and (2) doses of these magnitudes to the thyroid of a small child has some probability of producing cancer of the thyroid in later years.

the industry, with definite indication of opposition to formal siting regulations. AEC-R 2/20 contains excerpts of comments which the AEC received in writing together with comments made at meetings of the Technical Appraisal Task Force on Nuclear Power of the Edison Electrical Institute (EEI) on June 1, 1959, and the Atomic Industrial Forum on June 30, 1959.

14. In December, 1959, the General Manager appointed an Ad Hoc Committee to study the question of what the Commission can and should do at this time in the way of establishing definitive standards and criteria in the field of nuclear reactor safety. In a report to the General Manager dated September, 1960, the Committee recommended, "there be established rules which may of necessity involve some degree of arbitrariness, by which sites that would be considered acceptable for locations of reactors could be selected."

15. A draft of criteria along the lines of the proposed regulation was submitted to the ACRS for review and comments. A copy of that earlier draft is being circulated as AEC-R 2/22. The ACRS by letter to the Chairman, AEC, dated September 26, 1960 (Appendix "C-1") expressed the view that the proposed criteria could be developed into a useful contribution to nuclear safety studies but the criteria document should not be given the status of a Commission regulation. A similar recommendation, together with additional comments, was made by the ACRS in a letter of October 22, 1960 to Chairman McCone. (Appendix "C-2")

DISCUSSION

16. The primary objections of the ACRS (Appendix "C-2) to issuance of site criteria in the form of a regulation are concerns that:

a. Quantitative criteria established at this time in regulations would become so firm as to hamper unduly adaptation or modification to keep pace with changes that may prove desirable as the industry develops.

b. From the technical viewpoint, the simplification represented by the criteria, and the fixation by regulation of formulae such as those proposed for atmospheric dilution effects, accredit too great a validity to expressions that are at best approximations.

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c. Regulations with set numbers would be too restrictive and would deter efforts in nuclear safety progress toward a better set of limits.

d. The appearance of quantitative numbers in a Federal regulation would reduce the interest of the applicant in remaining alert for unforeseen disadvantages of a site and taking corrective action accordingly.

e. The correctness of the numbers which could be selected now cannot be proved by experimental or empirical data and, therefore, such numbers would give a false sense of positiveness which could not be supported under detailed scrutiny.

17. The proposed criteria (Appendix "D") establish as bench marks for site evaluation three characteristics distances for a reactor of any given power level: (1) an exclusion distance, (2) a (evacuation?) distance encompassing a surrounding zone of low population density, and (3) a distance to a defined population concentration. The criteria provide for evaluation of these bench mark distances in any individual case in accordance with the unique features and circumstances of that specific reactor project. The bench marks may be expressed in three different ways as shown in Annexes 1, 2 and 3 to Appendix "D". These alternate forms of presentation are included to assist in evaluation of the format in which such criteria might be published.

18. The first two bench mark distances and their corresponding dose limits as defined in the proposed regulation are as follows:

a. <u>Exclusion distance</u> - At this distance following the onset of the maximum credible accident the total radiation dose received by an individual in two hours would not exceed 25 rem whole body exposure or 300 rem to the thyroid from radioactive iodine exposure.

b. <u>Evacuation distance</u> - The greatest distance from the facility at which the total radiation dose received by an individual located at such distance and exposed during the whole course of the maximum credible accident to the radioactive cloud resulting from the accident would be 25 rem to the whole body, or 300 rem to the thyroid from radioactive iodine exposure.

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19. If one could be absolutely certain that no accidents greater than the maximum credible accident would occur, then the two distances specified above would provide reasonable protection to the public under all circumstances. There does exist, however, a theoretical possibility that substantially larger accidents conceivably could occur. It is believed prudent at present, when the practice of nuclear technology does not rest on a solid foundation of extended experience, to provide protection against the most serious consequences of such theoretically possible accidents. A third bench mark distance is, therefore, prescribed by which the reactor would be sufficiently removed from the nearest major concentration of people that no lethal exposures would occur in this population center even from an accident in which the containment is breached. The limit proposed for this third bench mark distance is defined in terms of possible radioactive effects under conditions of a contained maximum credible accident but represents the same distance that would insure no lethal doses in the event the containment is breached. The specification for this distance is:

> Population center distance - The distance from the facility at which the total radiation dose from the contained maximum credible accident received by an individual located at such a distance would be in the range of 50 to 100 rem to the thyroid from radioactive iodine exposure. It is fixed in the proposed regulation at 133-1/3% of the evacuation distance.

20. Provisions are made in the criteria for consideration of other relevant factors as well as the bench mark distances. The application of these criteria depends to a substantial degree on the subjective evaluative judgments of the person responsible for final approval of a reactor site. Thus adoption of these criteria will not provide fully objective procedures for site selection. Rather these procedures define bench mark distances as a beginning point in the evaluation process. This would be in

contrast to the methods which have been utilized to the present time. There has been no common point of departure and hence the entire process has depended upon subjective judgment.

21. The bench mark distance factors have been defined in the proposed regulation (Annex 1 to Appendix "D") in terms of integrated dose effects that might be experienced under the postulated accident. This method of presentation has the following advantages:

a. The potential radiations hazard expressed in integrated dose is the end form desired by the evaluator for judging the suitability of proposed sites.

b. Both the nuclear industry and the public think about nuclear hazards in terms of possible radiation doses. The criteria would thus be defined in terms likely to be best understood.

c. The position of the AEC would be clearly defined with respect to emergency dose limits that are now being used by much of the industry as reference limits for site selection and reactor plant design purposes.

22. The disadvantages to this form of presentation are:

a. The dose limits specified represent a certain degree of arbitrariness.

b. Limits on effluent releases from reactor installations during normal operations are currently specified in 10 CFR Part 20 in terms of nuclide concentrations. A simple comparison between allowable normal releases and possible releases under catastrophic conditions could not be made without some computation.

23. The same bench mark distances can be rewritten as shown in Annex 2 to Appendix "D" to express the distance factors in terms of the concentration of the predominant radioactive fission product that would contribute to the integrated dose at the bench mark distances. The advantages of defining the bench mark distances in terms of concentrations rather than dose limits are as follows:

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a. Allowable effluents from normal plant operation are set forth in 10 CFR Part 20 in terms of nuclide concentrations. Therefore, a certain degree of consistency would exist between the proposed new Part 51 and Part 20.

b. The concentration of the radioactive nuclides is the fundamental quantity derived from the atmospheric diffusion calculations and thereby results in some simplification of the calculational method that must be specified.

24. The disadvantages to this form of presentation are:

a. The method represents an over-simplification of the actual radiation effect at the specified points. The numerical value desired by the hazard evaluator is the integrated effect of the various nuclides that contribute radiation dose to a receptor. This integration in turn is a complex function of numerous factors such as the different decay rates of the nuclides released, the velocity at which they are transported, and the rate at which they might be deposited out during the transit period.

b. Defining the distances in terms of a concentration tends to mask the dose limits which are the basis for the concentration limits. One of the variables that has led to differences in calculations in the past has been the different conversion factors applied. Expressing distance factors in concentration limits will not eliminate this condition.

25. A third method of presenting the proposed criteria is shown by Annex 3 to Appendix "D". In this annex, the bench mark distance factors as a function of power level have been calculated and presented in the form of a table. The basis upon which the table has been computed has been omitted. The advantage of such a scheme is its simplicity. A principal disadvantage is that the fundamental bases for establishing the bench marks are hidden. Of course, those bases could be explained by press releases, speeches, etc., but the staff feels that the best place to explain them is in the regulation itself.

26. After consideration of the relative merits of the various ways in which the criteria might be expressed, it is the opinion of the staff of the Division of Licensing and Regulation that the bench mark calculations as presented in the form shown in

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Annex 1 to Appendix "D" (combined with a precalculated table) wherein the distance factors are defined in terms of reference dose limits, will best serve the interests of both the nuclear industry and the public and most clearly defines the basis upon which the AEC intends to evaluate proposed reactor locations.

27. The calculational methods set forth in the criteria represent one approach which can be taken in the current state of the art. In this approach, highly complex phenomena involving parameters which vary over wide ranges of values, depending on detailed conditions and assumptions, are reduced to manageable dimensions by simplifying assumptions, specifying that certain secondary factors are to be ignored, and arbitrarily fixing the values of certain key parameters. In utilizing this method, it should be recognized:

a. That there is a substantial degree of artificiality and arbitrariness involved.

b. That the results obtained are only approximations, sometimes relatively poor ones, to the result which would be obtained if the effects of the full play of all the variables and influencing factors could be recognized - an impossibility in the present state of the art.

c. That the net effect of the assumptions and approximations is believed to give more conservative results than would be the case if more accurate calculations could be made. Further details on the conservatism involved are described in Appendix "B".

Justification for criteria issuance in the form proposed is not upon its technical exactness but upon the value of having defined the basis upon which the AEC approaches judgments on reactor site suitability at this time.

28. As an indication of what might be expected from the application of the proposed bench marks to the site selection process, the bench marks were applied to nineteen reactor projects that have been proposed or are currently authorized for construction. The results are tabulated in Appendix "E".

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APPENDIX "B"

CONSERVATISMS IN THE ASSUMPTIONS AND FACTORS USED IN CALCULATING THE CONSEQUENCES OF THE MAXIMUM CREDIBLE ACCIDENT

1. The probability and consequences of catastrophic reactor accidents have been the subject of widespread interest and study since the earliest days of reactor development. To date, however, the technology has not progressed to the point where it is possible to assign quantitative numbers to all the significant factors relative to safety or to predict with surety the probabilities of malfunctioning of engineering features of plant design under all operating conditions that might exist. There is rather general agreement, however, as expressed in the Brookhaven Report (AEC Report WASH-740, Theoretical Possibilities and Consequences of Major Accidents in Large Nuclear Power Plants), that the probability of a major accident in reactor plants as we know them today is exceedingly small. The following is quoted from the report:

> "As to the probabilities of major reactor accidents, some experts believe that numerical estimates of a quantity so vague and uncertain as the likelihood of occurrence of major reactor accidents has no meaning. They decline to express their feeling about this probability in numbers. Others, though admitting similar uncertainty, nevertheless, ventured to express their opinions in numerical terms.... However, whether numerically expressed or not, there was no disagreement in the opinion that the probability of major reactor accidents is exceedingly low."

2. This low probability of occurrence is due to both the inherently safe features of reactors and the safeguards that have been engineered into the plants as a part of deliberate and planned effort to insure safety.

3. The conservatism reflected in the reactor plants is revealed through the analytical technique of postulating a severe accident condition and then evaluating the ability of the plant to remain under control and, through the safeguards

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provided, including location, prevent or minimize the effects of release of hazardous radioactive effluents. Whereas the exact probability of a major release cannot be predicted, it is possible to arrive at a judgment on site suitability through analysis of the conservatism reflected both in design and the assumptions made in calculating the consequences of a major accident. This in brief is the general approach that has been used by the AEC and the ACRS to arrive at their judgments on applications for construction permits.

4. The "maximum credible accident" is defined as that accident, usually an imaginatively postulated one, which would result in the most hazardous release of fission products, the ? potential hazard from this accident would not be exceeded by that of any other accident whose occurrence during the lifetime of the facility would appear to be credible.

5. For pressurized and boiling water reactors, for example, the maximum credible accident has been postulated as the complete loss of coolant upon complete rupture of a major pipe, with consequent expansion of the coolant as flashing steam, meltdown of the fuel and partial release of the fission product inventory to the atmosphere of the reactor building.

6. Power and testing reactors presently being operated or under construction near inhabited areas, pursuant to licenses issued by the Commission, are enclosed within external containment vessels. This outer barrier to fission product release to the atmosphere has within its enclosure all or a substantial part of the primary plant coolant piping systems representing an inner barrier. Cladding on the fuel provides an additional barrier that acts as a retaining "can" for the fissionable material and the fission products formed. Thus, gross release of fission products to the atmosphere would only occur after the breeching of two inner barriers: the fuel

cladding and the primary system, and then the external "barrier of last resort," the containment building.

7. The manner by which this might be initiated must follow one of two processes. First, through uncontrolled energy release to the confined coolant to produce pressure enough to rupture the coolant piping; or through mechanical failure of the piping or pressure retaining barrier. In either case loss of the coolant would set the stage for possible fuel meltdown from the decay nuclear heat.

8. The rupture of the coolant system from high internal pressurss due to uncontrolled internal heat generation requires that:

(1) Reactivity control mechanisms fail to function, and

(2) High-pressure relief systems fail to perform,

(3) Pressures exceed rupture limits of the piping material.

These prior failures need not occur for the case of a <u>spontaneous</u> pipe rupture. However, for such a case, the assumption of a complete shear of a pipe represents an extremely unlikely event. Nevertheless, assuming that such a break should occur and coolant is lost, fuel melting requires that:

a. Decay heat is sufficient to increase fuel temperature to the melting point;

b. Safeguard systems provided to flood or spray the core with water are either inoperative or insufficient to keep fuel temperatures from rising.

9. Despite such safeguards as those described above, if a major release of fission products to the environment should occur, estimations of the exposure doses which might result to persons offsite are extremely difficult to make because of the complex and interwoven technical effects involved. Although the amount of each kind of radioactive material present in a reactor system can be estimated fairly closely, as a function of the power level history, how much of this material would be

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released as a result of an accident is highly unpredictable. Quantities in the order of 10 - 30% of the total inventory have been assumed in the past. Experimental data would indicate these values to be conservative but the exact release can vary so much from reactor system to system and with the detailed nature of an accident that the exact degree of conservatism is not known. Further, there is a multiplicity of possible patterns of atmospheric dispersal whereby these radioactive materials can be transported to areas beyond the site boundary and those patterns can vary markedly from one reactor location to another.

10. In accidents of the "maximum credible" type, the radioactive materials, along with erosion and corrosion products, first would be dispersed in the coolant through melting or rupture of fuel elements, then find passage to the outer containment barrier through breeches in the coolant system. On breeching, the further expansion to a larger volume and a lower pressure in the containment vessel results in steam, in addition to the gaseous fission products, and production of aerosols as well as miscellaneous sizes of particulate matters. Some ejected materials may conceivably burn on contact with air, thus increasing the volatiles and fractions of smaller particles. At the same time, a certain amount of fallout within the reactor building or containment structure might be expected as well as condensation of the steam upon contact with cooler surfaces. The fallout is complicated by conversion of normally gaseous fission products into solids by decay, and condensation of volatiles by cooling. Fallout by diffusion and settling process under gravity is complicated by the agitations of turbulence and convection. Superimposed on these factors is the radioactive decay resulting in reduction of source strength with time by conversion to more stable isotopes. All

[redacted]

these factors pose a very difficult problem if one attempts to determine with any exactness the radioactive content of the air which leaks out of the final barrier (containment vessel).

11. The end objective of estimating this radioactive load within this final barrier is to attain a starting point for calculating the radiation hazard to those in the surrounding environs. For those in close proximity, this container of radioactivity represents a source of direct gamma radiation, attenuated by such factors as the structural shielding, distance, time decay and shielding by the topography. For those at more distant points, the transport by air of the materials leaking from the containment vessel becomes determining. For air transport, factors such as the nature of the material leaking from the containment vessel, release height, particle deposition with distance, wind direction, speed and variability, and air temperature gradients become important, and many of these are a function of the area in which the reactor is located.

12. It is from this complexity of interwoven technical parameters that criteria for use in the selection of sites has been formulated. While these criteria represent a considerable simplification of the many complex phenomena involved, they represent the same very conservative approach to site selection that has characterized such evaluations in the past. The fundamental assumptions upon which the proposed bench mark distances are based with estimates of the degree of conservatism represented in each case are as follows:

a. Experts agree and experience to date, though limited, confirms that there is only an exceedingly small probability of a serious accident in reactors approved or likely to be approved for construction. The probability is still lower for an accident in which significant amounts of fission products are released into the confined primary coolant system; and yet a great deal lower for accidents which would release significant quantities of radioactivity from the primary system into the reactor building.

was 75% nobles 25% halogens 1% solids

b. It is assumed that the maximum credible accident will release into the reactor building 100% of the noble gases, 50% of the halogens and 1% of the solids in the fission product inventory. This is approximately equal to 15% of the total fission product inventory. (The other 85% remain trapped within the fuel matrix or the plant primary system.)

c. The release of radioactivity from the reactor building to the environment shall be considered to occur at a leak rate of 0.1% per day. It is assumed that the leakage and pressure conditions persist throughout the effective course of the accident, which for practical purposes, is until the iodine activity has decayed away.

The maximum pressure within the reactor building and the leakage would of course decrease with time as the steam condenses from contact with cooling surfaces. By assuming no change in leak rate as a function of pressure drop, a conservative factor of at least 5 - 10 is introduced into final off-site dose calculations.

d. 50% settling of particles in the containment vessel is assumed in the bench mark criterion but credit has not been taken for the effects of washdown or filtering from protective safeguards such as cooling sprays and internal air recirculating system.

new

It is estimated that settling could give an effect of 3 - 10 reduction in the end result. Washdown features and filtering networks could provide additional reduction factors of 10 - 1000.

e. Atmospheric dispersion of material from the reactor building is assumed to occur according to a relationship developed by O. G. Sutton involving meteorological factors of wind velocity, atmospheric stability and diffusion parameters.

This relationship is representative of the current state-of-the-art for calculating downwind concentrations of dispersed material from a source, though there are other more complex relationships believed to be somewhat more accurate - and less conservative. It has been estimated that the use of the more accurate equations might result in reduction in calculated effects by 3 at distances in the order of 3 miles and a factor greater than 3 at 10 miles.

f. The bench marks assume no shift in wind direction for the duration of the accident.

The effect of assuming wind variability depends upon the pressure reduction rate within the containment vessel. Reductions in the order of 2 - 50 might be realized through wind direction shifts. Wind meandering from any one centerline direction might also result in a reduction factor of approximately 3.

Radioactive decay? I to shield I for city distance + other - hold up time in the containment

was not worst 2% (?)

g. Atmospheric dispersion is assumed to be under inversion type weather conditions. For weather conditions which exist for 75% or so of time at most sites, the atmospheric dispersion conditions would be more favorable, by factors of 5 - 1000.

h. No ground deposition (particulate fallout) is assumed for the evacuation distance.

Did get credit for city distance

Deposition during cloud travel could reduce the evacuation distance by factors of 2 - 5.

Thus, there is exceedingly high probability that, even if a maximum credible accident should occur, the resulting exposure doses would be many times lower than those calculated by the proposed bench mark calculations.

13. On the other hand, it must always be remembered that there are potential, conceivable accidents which would involve larger fission product releases than those assumed to be released in the maximum credible accident, and conceivably the consequences could be more hazardous to people. This, and other potentially more hazardous factors than those represented by the proposed site criteria, include:

a. Total radioactivity releases could theoretically be up to six times as large as those assumed.

b. Release of long-lived fission products could theoretically be up to 99 times as large as those assumed. This would have far ranging effects on bone dose exposures and on long term contamination of ground areas.

c. The weather conditions could be worse than those assumed, over a small percentage of the time, increasing exposure doses by a factor of 10 or more.

[redacted]

APPENDIX "C-1"

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
UNITED STATES ATOMIC ENERGY COMMISSION
WASHINGTON 25, D. C.

September 26, 1960

Honorable John A. McCone
Chairman
U.S. Atomic Energy Commission
Washington 25, D. C.

Subject: CRITERIA FOR EVALUATION OF REACTOR SITES

Dear Mr. McCone:

This is with reference to Mr. Finan's letter to me under date of September 21, 1960, in which the Advisory Committee on Reactor Safeguards is requested to transmit comments to you regarding a draft of criteria for the evaluation of sites for power and testing reactors proposed by the Division of Licensing and Regulation.

While the Committee believes that the present document could be developed into a useful technical contribution to reactor safety studies, there are a number of reasons why we cannot recommend that it be given the status of a Commission regulation.

We are sending you in the near future a memorandum on site criteria which sets forth the Committee's views on this matter.

Sincerely yours,

/s/

Leslie Silverman
Chairman

16—66209-1 GPO

APPENDIX "C-2"

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
UNITED STATES ATOMIC ENERGY COMMISSION
WASHINGTON 25, D. C.

October 22, 1960

Honorable John A. McCone
Chairman
U. S. Atomic Energy Commission
Washington 25, D. C.

Subject: REACTOR SITE CRITERIA

Dear Mr. McCone:

You have asked that we supply you with criteria which could be used for judging the adequacy of proposed sites for reactors. The Advisory Committee on Reactor Safeguards has devoted considerable time to this problem. A large part of our delay in submitting site criteria stems from the fact that we believe it is premature to establish quantitative limits on the variables involved in site evaluations - especially if such limits will appear in Federal regulations, or otherwise be announced as Commission policy. We recognize that the correctness of the numbers which could be selected now cannot be proved by experimental or empirical data, and, therefore, these numbers would give a false sense of positiveness which could not be supported upon detailed scrutiny. Numbers chosen now will be expected to change as more information develops. For example, a quantitative calculation of dosage must include some estimate of the fraction of the total fission product inventory which may be air-borne. This fraction is currently under experimental examination and the estimate may be subject to change.

The Committee believes that the officially endorsed numbers could stifle progress toward a better selection of numbers. The ideas and interpretations from applicants themselves have played a major part in the formulation of the current bases for site evaluation. It would be a significant loss to stop the flow of new ideas from the applicants. The Committee also believes that it is possible that the appearance of quantitative numbers in a Federal regulation or policy statement will reduce the continual

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awareness of the applicant that he has assumed a responsibility to be alert to and to act on unforeseen disadvantages of a site even after the site has been approved. The Committee, therefore, advises that a quantitative statement of site criteria not be included in Federal regulations.

These comments do not mean that the ACRS has no bases for judging the adequacy of sites. They merely emphasize that site selection is still largely a matter of judgment. Inasmuch as the ACRS has been making site and reactor evaluations, it may be helpful to review the framework on which these judgments are being made. It is a prerequisite, of course, that the reactor be carefully and competently designed, constructed, and operated. It should be inspected during all these stages in a manner to assure preservation of the intended protection of the public. Also, these factors are applicable only to those reactors on which experience has been developed. Reactors which are novel in design, unproven as prototypes, or which do not have adequate theoretical and experimental or pilot plant experience belong at isolated sites - the degree of isolation required depending on the amount of experience which exists.

Our site evaluations stem from several concepts. These are overlapping, but not conflicting:

1) Everyone off-site must have a reasonably good chance of not being seriously hurt if an unlikely but credible reactor accident should occur.

2) The exposure of a large segment of society in terms of integrated man-rems should not be such as to cause a significant shortening of the average individual lifetime or a significant genetic damage or a significant increase in leukemia - should a credible reactor accident occur.

3) There should be an advantage to society resulting from locating a plant at the proposed site rather than in a more isolated area.

4) Even if the most serious accident possible (not normally considered credible) should occur, the numbers of people killed should not be catastrophic.

Incidentally, the concept has been proposed by others that the damage to people from reactor accidents can be accepted if it is no greater than that experienced in other industries. We reject this suggestion as premature, and follow rather the concept that the consequences of reactor accidents must be less than this. The reasons for this rejection are twofold: First,

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we do not have sufficient information on the probability of reactor accidents to make use of this concept in site evaluations. We do use, of course, the fact that the probability of a serious accident is very low. Second, we recognize that the atomic power business has not yet reached the status of supplying an economic need in a manner similar to that of more mature industries; and, therefore, arguments for taking conventional risks for the greater good of the public are somewhat weak. At the same time, we do not want to imply that the restrictions placed on site locations during the developmental period of atomic power will necessarily be carried over to the period of maturity of the atomic power industry.

The reduction of these concepts to a judgment as to the adequacy of a proposed site requires further logic and the introduction of some numerical estimates. We believe that the searching analysis which is necessary at this stage should be done independently by the owner of the reactor, using the characteristics which are peculiar to his site and to his specific reactor. This step, we believe, is essential in developing his continuing alertness to his responsibility to the community surrounding the site. However, in Committee deliberation, we balance his analysis against a generalized accident which serves as a reference point from which we can better understand the analysis submitted by the applicant.

Our generalized accident analysis assumes that a serious accident has occurred and predicts in rough terms the consequences of such an accident. It is obvious that the generalized accident is an arbitrary artifact subject to change and has value only so far as it aids judgment. As a matter of fact, for certain reactors and conditions judgment will indicate that the generalized accident is too severe. In the generalized accident, we must make numerical assumptions as to the amount, type and rate of radioactivity release (the source term), the dispersal of the radioactivity in the air and in the hydrosphere, and the effect of this radioactivity on people.

Source Term

An arbitrary accident is assumed to occur which results in the release of fission products into the outermost building or containment shell. About 100% of the total inventory of noble gases, 50% of the halogens, and 1% of the non-volatile products are assumed to be so released. It is then assumed that this mixture leaks out of the outermost barrier at a rate defined by the designed and confirmed leak rate. The reasoning back of this source term is admittedly loose. It stems primarily from a present inability to be convinced that coolant cannot be lost somehow from the reactor core, either by spontaneous fracture of some element in the primary system a fracture caused by maloperation (instrumental or human) of the control rods. Admittedly, this assumed source term is large, but it thereby affords a factor of safety. In some cases it is justifiable to reduce this source term. It is also tacitly assumed that in this accident the outermost barrier will not be breached. The logic behind this assumption is that we require all of the components restraining the pressure of the primary system to be operating at temperatures above their nil-ductility temperature. We are, therefore, more confident, but not certain, that failure will occur by tearing rather than by brittle fracture and that the probability of ejection of missiles which penetrate the outermost barrier is low. The necessary supporting structures and shielding also protect against missile damage.

Dispersal of the Radioactivity

1) Meteorology

We assume a dilution of air-borne activity using atmospheric diffusion parameters which reflect poor, rather than average, meteorological conditions. Choice of specific parameter values follows from a survey of meteorological conditions expected to apply at the site, primarily wind and stability distributions. To analyze the generalized accident, we use the standard diffusion calculation methodology outlined, for example, in AECU-3066 and WASH-740. The atmospheric diffusion phenomena is the subject of active research, and new results can be expected to firm up and improve the present methods, although we do not anticipate major revisions in this area.

2) Hydrology

Considerations of hydrology are based on characteristics of surface and sub-surface flow as they are related to the possible release of contaminated liquids to the off-site environment. Thus, the rate and volume of surface flow and the possible presence or absence of absorbing barriers of soil between the reactor complex and important underground aquifers should be taken into consideration. These factors must be favorable for restraining the flow of radioactive materials in case of accident. Design factors, including the capability of providing adequate hold-up in the event of adverse hydrology, are also significant.

Effect of Radioactivity on People

The upper limit to the exposure to a member of the public in the generalized accident should be no higher than the maximum once-in-a-lifetime emergency dose. Such a level has not been established by AEC. We are arbitrarily using a figure of about 25 r whole body

[redacted]

or equivalent integrated dose for this level. This figure is mentioned in Handbook 59 of the National Bureau of Standards, pages 69-70. Since the iodine dose is often controlling, we are tentatively considering a thyroid dose limitation of 200-300 rads. The dosage so far mentioned refers to limits to people when the people are considered as independent individuals. We believe that it is essential that the Atomic Energy Commission attempt to confirm through its staff or its advisors in this field that this suggested value of 25 r whole body or equivalent is without significant biological effect on the individuals who might be subjected to this dose from the generalized accident.

When large numbers of individuals are exposed to radiation, another limit also exists because of genetic effects and because of the statistical nature of induced leukemia and the shortening of the life span. The limits of exposure to large groups of people are better expressed in terms of integrated man-rems. We are considering using a figure of 4×10^6 man-rems for this limit for the people who might be exposed to radiation doses falling between 1 and 25 rems. This figure of 4×10^6 man-rems is roughly equal to the dose received from natural background by a million people during their reproductive lifetime.

The implication of these numbers is this. About a reactor site, there should be an exclusion radius in which no one resides. Surrounding this, there should be a region of low population density, so low that individuals can be evacuated if the need arises in a time which will prevent their receiving more than a dose of 25 r. Beyond this evacuation area, there should be no cities (above 10,000 to 20,000 population) sufficiently close so that the individuals in these cities might receive more than the lower of the following: (1) 4×10^6 man-rems in the generalized accident, and (2) 200 rems under the extremely improbable accident in which the outermost barrier fails completely to restrain all of the radioactivity of the generalized accident.

The Committee wishes to emphasize again that the numbers which have been used in discussion of the generalized accident should not be formalized into regulations or Commission policy. The Committee wishes to acknowledge the help it has received from the Hazards Evaluation Branch in this matter and suggests that these individuals be encouraged to present as technical papers, but not as regulations, a complete description of their working approach to making judgments on the adequacy of proposed reactor sites. Such a paper, of course, would have the status of the opinion of an informed technical individual, but would not imply Committee approval, nor would it have the rigidity of a Commission policy statement.

Sincerely yours,

Leslie Silverman

Leslie Silverman
Chairman

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~~APPENDIX "D"~~

ATOMIC ENERGY COMMISSION

/10 CFR Part 51/

REACTOR SITE CRITERIA

Notice of Proposed ~~Rule Making~~

Statement of Considerations. On May 23, 1959 the Atomic Energy Commission published in the Federal Register a Notice of Proposed Rule Making that set forth general criteria for the evaluation of proposed sites for power and testing reactors. Many comments were received from interested persons reflecting, generally, opposition to the publication of site criteria, as an AEC regulation, both because such a regulation would, to some extent, incorporate arbitrary limitations and because it appeared that in view of the lack of available experimental and empirical data specific criteria could not be established.

Judgment of suitability of a reactor site for a nuclear plant is a complex task. In addition to normal factors considered for any industrial activity, the possibility of release of radioactive effluents requires that particular attention be paid to physical characteristics of the site, which may cause an incident or may be of significant importance in increasing or decreasing the hazard resulting from an incident Moreover, inherent or engineered design features of the reactor are of paramount importance in determining the possibility and consequences of any release of radioactive effluents. All these factors must be considered in determining whether location of a proposed reactor at any specific site would create an undue hazard to surrounding population.

Recognizing that it is not possible at the present time to define site criteria with sufficient definiteness to

[redacted]

eliminate the exercise of agency judgment, the proposed ~~rule~~ guides set forth below ~~is~~ are designed primarily to identify a number of factors considered by the Commission and the general criteria which are utilized as guides in evaluating proposed sites. Through the use of certain assumptions and ~~general~~ suggested calculational techniques set forth in Appendix "A", the proposed ~~rule~~ guide also attempts to establish a common starting point from which location factors can be assessed by the Commission, the applicant and other interested parties.

The proposed rule stems from the premise that a reactor should be so designed and located that the accident having a credible possibility of occurrence during the lifetime of the reactor, which would result in the most hazardous release of fission products (the maximum credible accident), would not result in undue hazard to the health and safety of the public. In assessing the potential hazard from the maximum credible accident, it is useful to consider its possible effect on three areas surrounding the reactor:

(1) The exclusion area upon which the reactor is located, an area access to which is under the direct control of the operator;

(2) The evacuation area surrounding the exclusion area, and area from which residents could be evacuated before any substantial radiological exposure could occur in the event of a reactor accident; and

(3) Nearby population centers, areas of high population density, evacuation from which probably would be neither desirable nor feasible.

The proposed rule describes a suggested calculational procedure for establishing references, or bench marks, based on power level, for use as a beginning point in site evaluation for a particular reactor. For the purpose of establishing bench marks only the calculational procedure assumes that all reactors are alike except for power level and that all site conditions are alike. The bench marks are:

[redacted]

(1) A bench mark exclusion area of such size that an individual located at any point on its boundary for two hours immediately following onset of the maximum credible accident would receive a total radiation dose to the whole body of 25 rem or a total radiation dose of 300 rem to the thyroid from iodine exposure.

(2) A bench mark evacuation area of such size that an individual located at any point on its outer boundary who is exposed to the radioactive cloud resulting from the accident (during the entire period of its passage) would receive a total radiation dose to the whole body of 25 rem or a total radiation dose of 300 rem to the thyroid from iodine exposure.

(3) A bench mark population center distance of 133 1/3% of the distance from the reactor to the nearest population center of more than 25,000 residents. An individual at this distance who is exposed to the radioactive cloud (during the entire period of its passage) would receive a total radiation dose in the range of 50 to 100 rems to the thyroid from iodine exposure.

The bench mark areas and distances can ~~are to~~ be obtained through use of the table on the calculational techniques contained in Appendix "A", which are designed to incorporate conservative factors and assumptions.

The whole body dose of 25 rem referred to in the bench ~~marks~~ corresponds to the once in a lifetime accidental or emergency dose for radiation workers which the NCRP recommends may be disregarded in the determination of their radiation exposure status. (See Addendum dated April 15, 1958 to NBS Handbook 59). The NCRP has not published a similar statement with respect to portions of the body, including doses to the thyroid from iodine exposure. For the purpose of establishing bench-mark areas and distances under the conditions assumed in the proposed guides ~~rule~~, the whole body dose of 25 rem and the 300 rem dose to the thyroid from iodine are believed to be conservative values.

As previously indicated, ~~these~~ bench marks are only a starting point in the evaluation of a proposed reactor location. The proposed guide ~~rule~~ specifies that the ~~Commission~~ will also consider

[redacted]

physical characteristics of the site, such as seismology, meteorology, hydrology, and geology; and characteristics of the reactor, such as maximum power level, proposed use, engineering safeguards, and unique design features. The over-all judgment is based on these features as well as the population density factors represented by the bench marks. Obviously, as specifically indicated in the proposed ~~rule~~ guide, the Commission may approve a proposed site which does not meet the bench marks or may disapprove a proposed site which does meet the bench marks.

Although approval or disapproval of a site will be evidenced by Commission action upon an application for a construction permit, the proposed ~~rule~~ guide provides that a preliminary report on site acceptability may be furnished by the Commission.

Notice is hereby given that adoption of the following ~~rule~~ guide is contemplated. All interested persons who desire to submit written comments and suggestions for consideration in connection with the proposed ~~rule~~ guide should send them to the Secretary, United States Atomic Energy Commission, Washington 25, D. C., Attention: Director, Division of Licensing and Regulation, within ~~ninety~~ five days after publication of this notice in the Federal Register.

[redacted]

(List of Section Headings)

AUTHORITY:

GENERAL PROVISIONS

§ 51.1 Purpose. ~~It is the purpose of the regulations in~~ this part ~~to~~ describes the criteria which guide the Commission in its evaluation of the suitability of proposed sites for power and testing reactors subject to Part 50 of this chapter. Because it is not possible to define such criteria with definite ness to eliminate the exercise of agency judgment in the evaluation of these sites, the ~~regulations set forth in~~ this part is ~~are~~ designed primarily to identify a number of factors considered by the Commission and the general criteria which are utilized as guides in approving or disapproving proposed sites.

§ 51.2 Scope. This part applies to applications filed under Part 50 of this chapter for construction permits and operating licenses for power and testing reactors.

§ 51.3 Definitions. As used in this part:

(a) "Exclusion area" means the area surrounding the reactor, access to which is under the full control of the reactor owner. This area may be traversed by a highway or railroad, provided such highway or railroad is not so close to the facility as to interfere with normal operations, and provided appropriate and effective arrangements are made to control traffic on the highway or railroad to protect the public health and safety. Residence within the exclusion area shall be minimal and residents shall be subject to ready removal in case of necessity to minimize hazard. Activities unrelated to operation of the reactor may be permitted in an exclusion area provided that no significant hazards to the public health and safety will result from the location of the activity in the exclusion area.

[redacted]

b. "Evacuation area" means the area immediately surrounding the exclusion area which contains residents the total number of which is such that there is a reasonable probability that they could be evacuated from the area or other counter measures could be taken in the event of a maximum credible accident before receiving substantial radiation exposures. The Commission has not specified a permissible population density or total population within the evacuation area because it may vary from case to case. Whether a specific number of people can be evacuated from a specific area on a timely basis will depend on many factors such as location, number, and size of highways, scope and extent of advanced planning, and actual distribution of residents within the area.

c. "Population center distance" means the distance from the reactor to the nearest boundary of a population center containing more than 25000 residents.

d. "Maximum credible accident" means that accident having a credible possibility of occurrence during the lifetime of the reactor which would result in the most hazardous release of fission products.

e. "Power reactor" means a nuclear reactor of a type described in § § 50.21 (b) or 50.22 of Part 50 of this chapter designed to produce electrical or heat energy.

f. "Testing reactor" means a "testing facility" as defined in § 50.2 of Part 50 of this chapter.

§ 51.4 Interpretations. Except as specifically authorized by the Commission in writing, no interpretation of the meaning of the regulations in this part by any officer or employee of the Commission other than a written interpretation by the General Counsel will be recognized to be binding upon the Commission.

[redacted]

SITE EVALUATION FACTORS

§ 51.10 Factors to be Considered When Evaluating Sites. In determining the acceptability of a site for a power or testing reactor, the Commission will take the following factors into consideration:

(a) Population density and use characteristics of the site and its environs, including, among other things, the exclusion area, evacuation area and population center distance.

(b) Physical characteristics of the site, including, among other things, seismology, meteorology, geology and hydrology.

(c) Characteristics of the proposed reactor and its use.

§ 51.11 Application of Site Evaluation Factors. The method by which the Commission will evaluate the factors described in § 51.10 is as follows:

1. Bench Mark Areas and Distances. A bench mark exclusion area, a bench mark evacuation area, and a bench mark population center distance will be established for each reactor, a a by calculational procedure~~s~~ described in Appendix "A" of this part.

(i) The bench mark exclusion area is an exclusion area of such size that an individual located at any point on the exclusion area boundary for 2 hours immediately following the onset of the maximum credible accident would receive a total radiation dose to the whole body of 25 rem or a total radiation dose of 300 rem to the thyroid from radioactive iodine exposure.

(ii) The bench mark evacuation area is an evacuation area of such size that an individual who is located at any point on the outer boundary of the evacuation area and who is exposed to the radioactive cloud resulting from the maximum credible accident (during the entire period of the cloud's passage) would receive a total radiation dose to the whole body of 25 rem or a total radiation dose of 300 rem to the thyroid from radioactive iodine exposure.

(iii) The bench mark population center distance is 133 1/3 of the distance from the reactor to the outer boundary of the evacuation area.

2. Relation of Bench Mark Areas and Distances to Other Factors. The establishment of bench mark areas and distances is for preliminary guidance as a beginning point in site evaluation for a particular reactor. The calculational methods used in establishing the bench marks incorporate significant assumptions concerning matters which are not susceptible of proof by experimental or empirical data and do not take into account individual site characteristics or specific reactor characteristics. Thus the bench mark areas and distances are not determinative for any reactor site but must be considered along with other relevant information. The Commission may approve a reactor site which does not meet the bench mark areas and distances, and it may disapprove a site which does meet the bench mark areas and distances. For example:

(i) Where the design of a particular facility incorporates extensive and well proven engineering safeguards or there are favorable features of the site or surrounding area, a proposed site may be approved even though its areas and distances are less than the bench mark areas and distances.

(ii) A site which meets the bench mark areas and distances may be disapproved for a proposed facility if the site or surrounding area has unfavorable features or if the proposed facility has unproven features.

(iii) In considering the suitability of a site for a proposed power or testing reactor, the Commission will consider the earthquake history of the site and its environs. The design for the facility should conform to accepted building codes or standards for areas having equivalent earthquake histories. No facility should be located closer than 1/2 mile from the surface location of a known active earthquake fault.

(iv) In considering the suitability of a site for a proposed power or testing reactor, the Commission will consider special meteorological conditions at the site and in the surrounding area.

(v) In considering the suitability of a site for a proposed power or testing reactor, the Commission will consider geological and hydrological characteristics of the proposed site which might have a bearing on the consequences of an escape of radioactive material from the facility. Power and testing reactors should not be located at sites where radioactive liquid effluents from an accident might flow readily into nearby streams or rivers or might find ready access to underground water tables.

(vi) Where some particularly unfavorable feature of the site exists, such that one or more of the criteria specified in paragraphs (i) to (v) of this paragraph are not met, the proposed site may nevertheless be found to be acceptable if the design of the facility includes appropriate and adequate compensating engineering safeguards.

[redacted]

(vii) In considering the suitability of a site for a proposed power or testing reactor, the Commission will consider proposed maximum power level; proposed use of the facility; the extent to which the design of the proposed facility incorporates extensive and well proven engineering standards; and the extent to which the reactor incorporates unique or unusual features having a significant bearing on the probability or consequences of accidental releases of radioactive material.

§ 51.20 Preliminary Review. Approval or disapproval of a proposed site will be evidenced by Commission action upon an application for a construction permit in accordance with applicable procedures and requirements under the regulations, ~~in this chapter~~. The Commission may, however, furnish a preliminary report as to the acceptability of a site proposed for a power or testing reactor prior to the filing and action upon an application for a construction permit.

[redacted]

[redacted]

See revised page

ANNEX 1 TO APPENDIX "D"

APPENDIX A

Calculation of Bench Mark Areas and Distances

1. On the basis of specified calculation methods and assigned values of parameters involved, bench mark areas and distances for reactors of various power levels have been determined and are listed in the following table:

Power Level (Thermal Megawatt)	Exclusion Distance (Miles)	Evacuation Distance (Miles)	City Distance (Miles)
1500	.59	13.3	17.7
1200	.51	11.5	15.3
1000	.42	10	13.3
900	.41	9.2	12.3
800	.39	8.4	11.2
700	.35	8.0	10.7
600	.32	7.1	9.5
500	.28	6.2	8.3
400	.25	5.2	6.9
300	.23	4.3	5.7
200	.21	3.5	4.7
100	.18	2.2	2.9
50	.15	1.4	1.9
10	0.8	.5	.7

2. This table has been based upon the following assumptions:

a. The maximum credible accident will release to the atmosphere of the reactor building 100% of the noble gases, 50% of the halogens and 1% of the solids in the fission product inventory. This release is equal to 15.8% of the total radioactivity of the fission product inventory. Of the 50% of the halogens released, one-half is assumed to condense out on the internal surfaces of the reactor building or adhere to internal components.

b. The release of radioactivity from the reactor building to the environment occurs at a leak rate of 0.1% per day of the atmosphere within the building and the leakage rate persists throughout the effective course of the accident which, for practical purposes, is until the iodine activity has decayed away.

[redacted]

[redacted]

c. In calculating the doses which determine the size of the bench mark areas, radioactivity decay in the usual pattern has been assumed to occur during the time fission products are contained within the reactor building. No decay was assumed during the transit time after release from the reactor building.

decay was allowed for I during transit to exclusion boundary

d. No ground deposition of the radioactive materials that leak from the reactor building was assumed.

was this permitted for city dist

e. The atmospheric dispersion of material leaking from the reactor building was assumed to occur according to the following relationship:

$$X = \frac{20}{\pi u C_y C_z d^{2-n}}$$

effect initial time dilution since no deposition allowed

where Q is rate of release of radioactivity from the containment vessel, the ("source term,"):

X is the atmospheric concentration of radioactivity at distance d from the reactor

u is the wind velocity

n is the atmospheric stability parameter

C_y and C_z are horizontal and vertical diffusion parameters resp.

π is a constant 3.1416.

f. Meteorological conditions of atmospheric dispersion were assumed to be those which are characteristic of the average "worst" (most favorable) weather conditions for average meteorological regimes over the country. For the purposes of these calculations, the parameters used in the equation in section e. above had values as follows:

$u = 1$ m/sec; $C_y = 0.40$; $C_z = 0.07$; $n = 0.5$

g. The isotopes of iodine were assumed to be controlling for the evacuation and city distances. The evacuation distance results from integrating the effects of iodine 131 through 135. The city distance equals the evacuation distance increased by a factor of one-third.

 Revised

~~e.g.~~ I^{131}

rel rate — Exclusion

$26(10)^3\ c/MW \times \underset{\text{released}}{.25} \times \frac{.001 \times 24}{24} = .54$ c/min

$26(10)^3\ c/MW \times \underset{\text{released}}{.25} \times .001 \quad \frac{8\ \text{days}}{.693} = 76.4$ c/day

Generator

average life $= \tau = \frac{1}{\lambda} =$

Q i c/[illegible]

[redacted]

h. The source strength for each iodine isotope was calculated to be as follows:

Isotope	Exclusion Q (curies/megawatt)	Evacuation Q (curies/megawatt)
I^{131}	.48	76.5
I^{132}	.55	1.44
I^{133}	.77	1.82
I^{134}	.62	.91
I^{135}	.87	5.4

These source terms combine the effects of fission yield under equilibrium conditions, radioactive decay during the holdup time in the reactor building, and the release rate from the reactor building.

i. For the exclusion distance, doses from both direct gamma radiation and from iodine in the cloud escaping from the reactor building must be calculated and the distance established on the basis of the effect requiring the greater isolation.

j. In calculating the thyroid doses which result from exposure of an individual to an atmosphere containing concentrations of radioactive iodine, the following conversion factors were used to determine the dose received from breathing a concentration of one curie per cubic meter for one second:

Isotope	Dose (rem)
I^{131}	334
I^{132}	12.7
I^{133}	78.8
I^{134}	6.14
I^{135}	21.9

1.04 $\mu c/cc$ at BR 1×10^{7} cc

k. The whole body doses at the exclusion and evacuation distances due to direct gamma radiation from the fission products released into the reactor building in the maximum credible accident were derived from the following relationships:

Revised

was 330

$$D = 483 \frac{Be^{-ur}}{4\pi r^2} \int_0^t t^{-0.21} \, dt$$

where D is the exposure dose in roentgens per megawatt of reactor power

r is the distance in meters

B, the scattering factor, is equal to $(1 + ur + \frac{ur^2}{3})$

u is the air attenuation factor (0.01 for this calculation)

t is the exposure time in seconds.

In this formulation it was assumed that the shielding and building structures provided an attenuation factor of 10.

[redacted]

Annex 2 to Appendix "D"
Appendix A (alternate 1)

Calculation of Bench Mark Areas and Distances (concentration limits)

The calculational procedure to arrive at bench mark areas and distances defined in terms of concentration limits is basically the same as that shown in Annex 1. The table of bench mark distances would be identical but the explanation of the assumptions used in deriving the table would differ in the following ways:

a. The evacuation distances would be derived from the following relationship:

$$d^{2-n} = \frac{2\,Q}{\pi\, u\, Cy\, C_z\, X}$$

where:

d is the distance

Q is the rate of release of radioactivity from the reactor building

u is the wind velocity

n is the atmospheric stability parameter

Cy and C_z are horizontal and vertical diffusion parameters

π is the constant 3.1416

X is the concentration limit for iodine defining the bench mark distance

b. Iodine isotope 131 would be assumed to be controlling. The concentration limit X would be defined to reflect contributing effects of the other iodine isotopes.

c. Conversion of concentrations into doses as described in par. 2j of Annex 1 would not be required.

[redacted]

Annex 3 to Appendix "D"

Appendix A (Alternate 2)

Table of Bench Mark Areas and Distances

In establishing bench mark areas and distances the following table shall be used:

TABLE OF BENCH MARK LOCATION DISTANCES

Power Level (Thermal Megawatt)	Exclusion Distance (Miles)	Evacuation Distance (Miles)	City Distance (Miles)
1500	.59	13.3	17.7
1200	.51	11.5	15.3
1000	.42	10	13.3
900	.41	9.2	12.3
800	.39	8.4	11.2
700	.35	8.0	10.7
600	.32	7.1	9.5
500	.28	6.2	8.3
400	.25	5.2	6.9
300	.23	4.3	5.7
200	.21	3.5	4.7
100	.18	2.2	2.9
50	.15	1.4	1.9
10	.08	.5	.7

Note: This table represents a pre-calculation of the bench mark areas and distances precluding the need for reference in the regulation to either dose limits or concentration limits.

APPENDIX "E"

BENCH MARKS FOR SELECTED REACTORS

		Exclusion Area		Evacuation Area		Population Center Distance	
MWt	Reactor	Bench Mark Distance (miles)	Actual Distance (miles)	Bench Mark Distance (miles)	Actual Pop.Density in Bench Mark Area (people/sq.mi.)	Bench Mark Distance (miles)	Actual Distance (miles)
630	Dresden	.33	.5	7.4	38	9.9	14
585	Con. Ed.	.31	.3	7.0	403	9.4 was 9.2	17 was 1.3
485	Yankee	.28	.5	6.2	33	8.3	21
300	FRDC	.23	.75	4.5	24	6.1	7.5
270	PWR	.23	.4	4.2	298	5.7	7.5
240	Consumers	.22	.5	3.9	28	5.2	135
240	Hallam	.22	.25	3.9	10	5.2	17
203	Pathfinder	.21	.5	3.5	25	4.6	3.5
202	PG&E	.21	.25	3.5	172	4.6	3
200	ICBWR	.21	.2	3.5	86	4.6	10
115	Phila. Elec.	.19	.57	2.4	29	3.2	21
60	NASA	.16	.57	1.6	53	2.1	3
60	CVTR	.16	.5	1.6	12	2.1	25
60	Jamestown (Orig. site)	.16	.3	1.6	1200	2.1	0.5
60	Jamestown (New site)	.16	.3	1.6	66	2.1	2.4
58	Elk River	.16	.23	1.5	40	2.0	20
50	VBWR	.15	.4	1.4	23	1.9	15
48	Piqua	.15	.14	1.4	960	1.8	27
40	Pt. Loma	.14	.25	1.2	0	1.6	3

[redacted]