MASTER

Siting of Nuclear Facilities

SELECTIONS FROM *Nuclear Safety*

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PREPARED FOR THE U.S. NUCLEAR REGULATORY COMMISSION
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Nuclear Safety Information Center

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FOREWORD

The Nuclear Safety Information Center (NSIC), which was established in March 1963 at Oak Ridge National Laboratory, is principally supported by the U.S. Nuclear Regulatory Commission's Office of Nuclear Regulatory Research. Support is also provided by the Division of Reactor Development and Demonstration of the U.S. Energy Research and Development Administration. NSIC is a focal point for the collection, storage, evaluation, and dissemination of safety information to aid those concerned with the analysis, design, and operation of nuclear facilities. A system of key words is used to index the information cataloged by the Center. The title, author, installation, abstract, and key words for each document reviewed are recorded at the central computing facility in Oak Ridge. The references are cataloged according to the following categories:

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ABSTRACT

This report explores siting policy and practice for nuclear power plants as developed in the U.S. and abroad. Twenty-two articles from *Nuclear Safety* on this general topic are reprinted since they provide a valuable reference source. The appendices also include reprints of some relevant regulatory rules and guides on siting. Advantages and disadvantages of novel siting concepts such as underground containment, offshore siting, and nuclear energy parks are addressed. Other topics include site criteria, risk criteria, and nuclear ship criteria.

1. INTRODUCTION

Nuclear power plant siting practice has generally been to place plants in dispersed sites that are away from highly populated centers. However, the projected nuclear power capacity of 10^6 MW(e) or more in the next 25 years has forced planners to consider siting reactors nearer population centers and/or to consider possible innovative siting concepts. New concepts being explored include underground containment, offshore siting, and nuclear energy centers. The philosophy, background, and trends in siting have been reviewed frequently in *Nuclear Safety*.

Since the siting of nuclear power plants brings together all the various considerations related to the protection of the health and safety of the public, it is not surprising to find that a significant number of articles on siting have appeared in *Nuclear Safety* during its 16 years of publication. The most timely ones have been selected and are grouped in this report in sections entitled Site Criteria, Risk Criteria, Nuclear Ship Criteria, Novel Siting Concepts, Foreign Practice, and Meetings on Siting. A brief discussion of the subject matter to be found in each section follows.

Site Criteria

U.S. Reactor Site Criteria were officially established in May 1962 with their incorporation into the Code of Federal Regulations as Part 100 of Title 10.¹ Three years of intensive study by the USAEC [now the Nuclear Regulatory Commission (NRC)] preceded adoption of the criteria. During this time, proposed criteria were published twice in the *Federal Register* and comments were solicited. The criteria were extensively revised on both occasions, based upon comments received from interested parties which included industry, governmental agencies, and the general public. The criteria are published in this report as Appendix A.

A sample calculation of environmental and distance characteristics that appeared as an appendix to this draft criteria was published in a separate document (TID-14844).² This action was taken after strong objections arose over its inclusion as part of the official regulations. For the reader's convenience, the report is reprinted here as Appendix B. The contents of both the site criteria and TID-14844 are discussed in the first

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article in this section (p. 14), while the second article (p. 20) summarizes the comments received by AEC during public review of the draft criteria prior to their finalization.

The criteria and supporting document describe the calculation of certain distance factors relative to the separation distance between the reactor and nearby populations. An exclusion area is defined as one in which residence will normally be prohibited. Around this area is a low-population zone in which the inhabitants are subject to protective measures such as evacuation. A population-center distance is defined as the distance to the nearby boundary of a densely populated area containing more than 25,000 residents. The sizes of the exclusion area and the low-population zone are related to calculations based upon the design-basis accident. In addition, the population center distance is taken as $1 \frac{1}{3}$ the radius of the low-population zone.

The requirements for nuclear power plant siting are still those set out in 10 CFR 100. In fact, in over 13 years since the criteria became official, the first modification of wording was made in June 1975.³ The change more clearly defines the meaning of the term boundary. The clarification was made because of a recent decision of the U.S. Court of Appeals for the Seventh Circuit which interpreted Part 100 in a way that deemphasized the role of population distribution in nuclear plant siting. Boundary had been defined as "the distance from the reactor to the nearest boundary of a densely populated center containing more than about 25,000 residents." The clarification makes official the interpretation of the term boundary that emphasizes population density rather than the political or corporate boundaries of the "center containing more than 25,000 residents."

Although Seismic and Geologic Siting Criteria were added to Part 100 as Appendix A,⁴ no changes were made in the body of the criteria since its adoption except as noted in the preceding paragraph. One can logically ask why the prose in 10 CFR 100 remained unchanged in a period when other AEC regulations were undergoing frequent revision. Principally, it is due to the flexibility that was written into the criteria. Part 100 grouped the site evaluation characteristics considered by the AEC into the three broad categories as follows: 1. Characteristics of Reactor Design, 2. Population Density, and 3. Physical Characteristics of the Site (see

article by Price, p. 34), but it also contained the important principle of compensating safety features. This means that where unfavorable characteristics of a proposed site exist, the site may still be found acceptable if the reactor design includes compensating engineered safety features such as improved air-cleaning systems, cooling systems, containment designs, etc. The principle includes characteristics related both to population density and to physical characteristics such as meteorology, hydrology, or seismology. For example, nuclear reactors located near large centers of population have led the way in incorporating improved safety features into their design. Examples of physical characteristic compensation are also evident. Power reactors located in coastal areas have included provisions in their design to prevent malfunctions resulting from the high winds and waters associated with hurricanes and/or tsunamis. Power reactors in midwest locations have included additional features to protect against tornadoes. Designs in areas of high seismic activity include additional provisions to withstand the ground vibrations associated with earthquakes.

The compensating nature of safety features are analyzed in some detail in the Ergen (p. 23) and Culver (p. 30) articles. The second of these articles, published in 1968, concluded that factors of credit could be calculated for safety features in those reactor stations that did not meet the site distance criteria per se. A later study, published in 1973 by Piper and Heddleson (p. 37), sought to establish a relationship between the number of safety features included in a plant and site characteristics. They thought it peculiar that no meaningful relationship could be found. The lack of a correlation appeared to be due to two factors. First, it is the practice of reactor manufacturers to sell a more-or-less standard plant, regardless of the location. This practice tends to result in the maximum number of safety features on all plants. The second factor is the Regulatory practice of retrofitting new safety features to older plants when these features are judged to substantially improve the safety of the plant.

An interesting feature of the Piper-Heddleson article, which looked at current siting practice in relationship to population, was the cumulative population vs distance curves for nuclear power plants. These curves

suggest that the population distributions described initially by the Indian Point site and subsequently by both Indian Point and Zion have served as an upper limit on approved sites. Those sites with larger population distributions have either been withdrawn or strongly challenged. The curves of foreign reactor plants show development of a trend toward siting closer to population centers in Europe than in the U.S. This is apparently due to their lack of abundant fossil fuel, high power needs, and high population density. However, metropolitan siting does not seem to be a near-term option in the U.S. Safety reliability established through experience will evidently be necessary to start a trend toward "closer-in" siting than is now being approved.

While 10 CFR 100 states that "where very large cities are involved, a greater distance may be necessary because of total integrated population dose considerations," AEC did not provide specific guidance as to acceptable population characteristics for a power-plant site beyond the low population zone.

In 1973 the Regulatory Staff of AEC had considered issuing an interim regulatory guide, *Population Distribution Around Nuclear Power Plants*, to provide this guidance (see article, p. 46). As part of the process of developing this guide and other related working papers, discussions were held with a number of persons and groups, both inside and outside the nuclear industry, to obtain their views. The draft guide and related papers were disseminated in WASH-1308⁵ and publicly released⁶ in April 1974. The main thrust of the proposed guide would be to provide further guidance, from the standpoint of compliance with 10 CFR 100, as to low-population-density characteristics and to require a sufficient justification for sites where the population densities are higher. However, the population guide was not formally issued but only circulated for discussion. The Nuclear Regulatory Commission is considering revision of this draft, but at this time the requirements for siting are still those set out in 10 CFR 100.

Population distributions around a proposed nuclear power facility are considered primarily to assure that risks to the population as a result of any accidental releases of radioactivity are adequately taken into account in the selection of the site. Since the routine releases of radioactivity allowed under NRC regulations are extremely small, the risks to the

population from these releases, even assuming a very high population density around the facility, can be fairly characterized as negligible.

Under 10 CFR 100, applicants are required to postulate a large release of fission products within the reactor containment and, on the basis of acceptable assumptions for various design-basis accidents provided by the Regulatory Staff, to calculate off-site exposures from such an assumed release. These calculated exposures may not exceed exposure guidelines established in 10 CFR 100, and, owing to the resulting conservative plant design, the risk to any individual from an accident is small. Part 100 also specifies that special considerations may apply when large population centers are near proposed power-reactor sites because the large number of persons who potentially could be exposed to radioactivity in the event of an accident does present some small additional increment of risk to the total population. The use of the proposed guide should assure that, with reactors of current technology, sites that have high population densities are not used in the absence of sufficient justification.

The NRC recognizes that siting of reactors near highly populated centers presents some additional increment of risk over that presented by a low-population-density site. Therefore, the intent of the population guide is to encourage siting in low-population-density areas unless high-density areas offer significant environmental, economic, or other advantages.

The recent Regulatory Guide 4.7, *General Site Suitability Criteria for Nuclear Power Stations*, is reprinted as Appendix C. The suitability guide discusses the major site characteristics related to safety, public health, and environmental issues which the Regulatory Staff considers in determining the suitability of sites for reactor plants. A letter from the Advisory Committee on Reactor Safeguards on siting evaluation is reprinted as Appendix D.

Risk Criteria

Since no engineering plant or structure of any type is entirely risk free, the building of a nuclear reactor implies the acceptance of some finite degree of risk. The articles by Farmer (p. 54) and Meleis and Erdmann (p. 63) discuss a logical way of dealing with this situation by seeking to assess a wide spectrum of risks in a quantitative manner to provide some choice in the selection of reactor sites.

In the first article, Farmer suggests that quantitative standards are becoming increasingly necessary because reactors are needed strategic to electrical demands in many countries which have areas of high population densities. Substantial effort made in the United Kingdom and other countries to improve the economic performance of reactors has been accompanied by a corresponding effort in the study of a wide range of potential accidents in order to estimate the quantity of fission products that might be released. Farmer feels that an obsessive preoccupation with these studies has diverted attention from two fundamentally important considerations, namely, the likely frequency of accidents and the consequent risks to people.

He points out that the isotopes of iodine, particularly ^{131}I , carry a greater threat to health than any of the other fission products that might be released in a reactor accident, and the International Commission on Radiological Protection (ICRP) has published findings of the risk of developing thyroid cancer through the inhalation of radioactive iodine.⁷ He suggests a way to estimate the likelihood of releasing various quantities of ^{131}I from a particular reactor and thereby calculating the risk associated with its operation on any site. A boundary line or upper-limit concept is suggested based upon the quantity and frequency of ^{131}I release as a simple way of defining a criterion that can be used by designers and assessors. It does not, in itself, define a total limiting risk, since this requires the summation of all risks that may be presented in different ways by the multiple subdivision of the accident studies. In principle, these can be summed, but Farmer suggests that the practice of risk assessment should be allowed to develop without excessive complexity and within the simplest possible target.

The Meleis-Erdmann article examines the application of Farmer's limit-line criterion. The total individual mortality risk per year for persons living at the exclusion distance of the plant was calculated and then compared with present socially recorded risks. As a result, they suggest a modified limit line leading to a calculated total risk that lies within present acceptable limits. The consequences of accidents calculated for a number of existing nuclear plants with the original and the modified limit line are compared.

The sampling of the nuclear plant probability analyses indicates that a limit-line safety approach is feasible with presently designed plants. Moreover, the authors relate, very simply, the limit-line data to an integrated risk value by factoring in weather data and exclusion radii. An approach based on the modified limit-line concept provides a great deal of insight into power-plant behavior that is useful in plant design.

The accidents postulated in nuclear power plants require a whole sequence of failures of safety equipment. Recent quantitative work on the likelihood of these failures occurring has been performed in the Reactor Safety Study headed by Prof. Norman Rasmussen of the Massachusetts Institute of Technology. A report⁸ on the work, published in October 1975, found that realistic probabilities of these events are very small.

Selected references from the large body of literature on risk analysis have been published in a recent NSIC bibliography.⁹

Nuclear Ship Criteria

The two articles in this section relate to port visits of the nuclear merchant ship N.S. *Savannah*, the first (p. 70) on site or port criteria and the second (p. 73) on port surveys. The 10 CFR 100 criteria were specifically intended to reflect stationary reactor siting practice in this country. The criteria not only did not cover many of the new problems introduced by a mobile reactor but also as written would have specifically excluded nuclear shipping from most harbors. In view of the many unique differences between siting a stationary power reactor and berthing a nuclear merchant vessel, the ship site criteria set forth in some detail the safety feature requirements, the conditions for port entry, and the local port arrangements that must be made. Such conditions and arrangements implement the general objectives of the ship criteria and ensure their attainment.

Other constraints affect the choice of berth in addition to the reactor safety considerations. For example, the berth must be of suitable size and offer adequate draft alongside to accommodate the vessel; its facilities must be capable of handling the flow of cargo, passengers, and visitors; its location must permit ready and economical approach and egress for goods and personnel and at the same time provide easy access for emergency vessels and minimal interference with port traffic if an accident should require

sudden removal of the ship; and the site must be unoccupied and available for the period of the projected visit.

The N.S. *Savannah* port surveys were performed by a team comprising a ship's captain and a nuclear engineer. The pair commonly initiated a port survey by arranging a joint meeting with administrative and regulatory officials from appropriate local, state, and federal bodies. The officials were briefed concerning the berth requirements and emergency procedures, arrangements were made to examine the docks, preliminary channels for ship's communications were established, and an emergency alert list was compiled. After the docks and their environs were inspected and a suitable berth was secured and locally approved, operational details such as time of arrival, escort vessels, tugboat service, pilotage, and location of remote anchorage were resolved; all affected parties were advised; and a survey report was prepared.

Novel Siting Concepts

During the past few years, a number of novel siting ideas have been proposed. These have included underground containment, off-shore siting, and nuclear energy parks. The objective is to better isolate the reactor from the surroundings and relieve some environmental constraints. However, these concepts inevitably introduce other considerations, while at the same time making new siting space available.

There is considerable experience on underground construction in a cave or in excavated rock, particularly in Europe. The first article on this type of containment for nuclear power plants appeared in *Nuclear Safety* in 1961 (p. 77), and two other articles appeared during the past year (pp. 102 and 118). While the cost estimates of the earlier article are subject to escalation, the basic conclusions of the article are still valid. It was concluded that underground construction of reactors was feasible and, in some areas, possibly competitive with aboveground construction of a steel containment shell. Furthermore, it could be justified in some cases on the basis of invulnerability to military attack. Since a power supply is one of the first requirements for reconstruction after a disaster, the increase in cost over that of the aboveground plant could be considered as insurance against total disaster.

Crowley and associates (p. 87) made a broad examination of underground siting on the basis of technical feasibility, design, safety, and economics. Overall safety improvements are cited relative to immunity to surface phenomena, modified and lessened seismic loadings, an additional level of containment, and greater ability to cope with class 9 accidents. These improvements are also discussed by Yadigaroglu and Anderson (p. 104). However, siting of large commercial nuclear plants underground has not proliferated for various reasons, such as (1) no real shortage of surface sites up to the early 1970s, (2) ambiguity of the licensing requirements, and (3) an apparent economic penalty.

It can be concluded that current technology is adequate for placing a nuclear plant totally or partially underground. A thorough evaluation program will be required to clearly establish the economics of the concept. Although the concept can be employed at any location, it may be particularly applicable in regions of high population density and/or high seismic activity.

Offshore siting was discussed by Anderson (p. 87) and Yadigaroglu and Anderson (p. 104) as a possible solution to the problem of finding acceptable land sites for nuclear plants. Many of the problems associated with siting nuclear power plants could be alleviated by the construction of offshore stations. Fixed (artificial island) offshore stations offer a practical siting alternative. However, of far greater potential is the floating-station concept, since it offers the possibility of a greater choice of offshore locations, of improvements in safety through earthquake resistance and a large exclusion distance, and of cost reductions as a result of factory assembly. An additional accident mode that must be considered is that of ship collision. The major danger of such a collision is not the impact, which will be mitigated by the breakwater or other barriers, but rather the consequences of flammable or explosive cargo releases.

The probabilities of major radioactivity releases remain similar to those of land-based facilities; however, the consequences of such accidents are not equivalent. Core meltdown and foundation penetration in a land-based plant might result in loss of life and related property and environmental damage. The same accident in a facility floating on an ocean or

other body of water might have more serious consequences because the beneficial retention and filtering capability of the ground is lost. The immediate consequences of airborne radioactivity would be lessened since the water would retain some of the escaping gaseous or volatile fission products, notably the radioiodines. This creates a potential problem of damage to the ocean, which is of considerable concern. Consideration is being given to the inclusion of additional protective features, such as a core catcher, to function in case of a meltdown.

The current practice of siting reactors has resulted in a number of dispersed sites being developed throughout the country. While the present tendency has been to disperse reactors, as many as four units are being planned for a limited number of sites. Until the tendency for clustering accelerates, the average number of reactors per site will be small. Cope, in his article on nuclear energy centers (p. 118), suggests that one assume an average of three reactors per site, which would result in about 470 sites (almost four times the present number) by the year 2000, with the number increasing rapidly thereafter. This raises the question as to whether the proliferation of nuclear reactors over hundreds of sites throughout the country within the next 40 to 50 years should be our long-range practice. He suggests that an important step to improving the posture of nuclear energy would be to establish nuclear energy centers.

There are several potential advantages for clustered nuclear activities; the most important of these are as follows:

1. more reliable safeguarding of nuclear materials;
2. a higher degree of health and safety protection by concentration of resources, strong centralized management, and reduced shipments of radioactive materials;
3. reduced impact on the environment as compared with dispersed sites;
4. more efficient use of land and other resources;
5. establishment of a minimum number of permanent radioactive sites;
6. provisions for more efficient control of radioactive wastes generated within the center;
7. faster construction of nuclear power plants;
8. possible cost savings in construction through repetitive practices.

However, there are a number of constraints and uncertainties that must be solved before the concept of energy centers can be developed completely. The more obvious ones are

1. technical constraints, such as transmission limitation, climatological effects, etc;
2. potential for common-mode failures due to natural disturbances, or other occurrences;
3. institutional, financial, and other arrangements;
4. legislative, insurance, and tax restraints;
5. community impacts;
6. the time required initially for planning and approvals.

Foreign Practice

Although there are only two articles (pp. 127 and 134) devoted exclusively to foreign practice in the siting of nuclear power plants, there have been related discussions in papers that were primarily on other aspects of reactor siting. For instance, the article on "Siting Practice and Its Relation to Population" (p. 37) contains cumulative population vs distance curves for sites in Canada, West Germany, the United Kingdom, and Japan. Siting practice in the U.K. was discussed by Farmer in "Reactor Safety and Siting - A Proposed Risk Criterion" (p. 54).

G. Hake (p. 127) compared siting practices of Canada and the U.S. Siting conditions that apply to the Canadian Pickering and the U.S. Indian Point 2 and Oyster Creek stations were analyzed in a general way to show that in both countries the approval of these stations involved three similar conditions: (1) a 10^5 -order reduction in the total risk, (2) credit for safety features equivalent to a first- or second-order reduction of risk, and (3) integrated population doses of about 10^6 rems to the thyroid. Hake suggested that a requirement based on a population dose limit is likely to be more easily accommodated than one based on a population distance limit. He also suggested that in the practical siting situations likely to arise, the individual dose at the exclusion boundary would govern, and the population center distance criteria as presently defined in the U.S. regulations were therefore unnecessarily restrictive.

Ergen (p. 134) found that West German policies with respect to siting were similar to those in the U.S. The Germans are somewhat more liberal where differences in philosophy exist. This is not surprising, since the Germans have fewer alternatives to siting reactors in more densely populated areas. However, sites approved up to 1970 were not substantially more populated than those approved in England or America.

Meetings on Siting

Nuclear Safety undertakes to review major national and international meetings pertaining to the siting of nuclear facilities. Included here are reviews of four such meetings, two by the International Atomic Energy Agency, and one each by the Health Physics Society (U.S.) and the American Nuclear Society.

The first two meetings, sponsored by the International Atomic Energy Agency, were held in Bombay in 1963 (p. 137) and in Vienna in 1967 (p. 150). Topics covered in the first review were power reactor siting in the United Kingdom and other countries, siting for research reactors, consequences of activity release, and the effects of engineered safety features. The principal topics covered at the Vienna meeting were national practices in reactor siting, reactor siting considerations, containment, and release and transport of pollutants.

In 1970, the Health Physics Society sponsored a Symposium on Health Aspects of Nuclear Facility Siting in Idaho (p. 154). Forty-two papers were given in sessions dealing with general site-selection criteria for nuclear power reactors; siting evaluations for large accelerators; fuel, coolant, and fission product interactions; evaluation of environmental contamination around nuclear facilities; behavior of airborne radioactivity; radionuclides in biological systems; comparison of hazards from internal contamination; radiation standards; and public relations.

The most recent meeting reviewed was one on Nuclear Power-Plant Siting sponsored by the American Nuclear Society (ANS) in Portland during August 1974 (p. 162). It included sessions devoted to the government and siting, methodologies of site selection, special and technical issues, utility siting experience, and social and public acceptance issues. In the opinion

of the reviewers, the conference was unique among ANS meetings in that most papers recognized, at least implicitly, the growing importance of social impacts of power-plant siting. Previous siting conferences have concentrated almost exclusively on quantifiable engineering aspects, such as seismic or thermohydraulic effects.

Epilogue

This collection of articles comprises a valuable source book on the siting of nuclear reactors. It should serve as an introduction text for those who are not familiar with the subject and as a reference source for those who are already knowledgeable in the business. As for new developments, *Nuclear Safety* will continue to inform and stimulate its readers with articles on these topics in the future.

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2. SITE CRITERIA

2.1 Reactor Site Criteria

Wm. B. Cottrell*

[*Nucl. Safety*, 4(1): 11-16 (September 1962)]

The Atomic Energy Commission (AEC) has established criteria for use in the evaluation of proposed sites for stationary power and test reactors licensed by the AEC. The approved Site Criteria Guides were released in the *Federal Register* on Apr. 12, 1962, and will be incorporated in Title 10, *Code of Federal Regulations*, as Part 100, effective 30 days from the date of publication. The approval of these guides was preceded by more than three years of intensive study, during which time proposed guides appeared twice in the *Federal Register*^{1,2} for comments—which were profuse on both occasions^{3,4}—and following which the proposed guides were extensively revised. The last revision incorporated many of the comments, and the published document has been designated a guide for an interim period until enough experience can be accumulated with reactors to permit the writing of more definite standards.

The approved site criteria differ from their immediate predecessors in several significant points. A predominant point of objection was the inclusion of a sample calculation of environmental and distance characteristics as an appendix to the proposed guides on the grounds that (1) the definite numerical values used in the sample calculation gave an aspect of undue rigidity to the guides and (2) the example tended to emphasize unduly the concept of environmental isolation. This objection was resolved not only by deleting the sample calculation from the guide and placing it in a separate document⁵ but also by identifying the criteria as interim measures and by stating that the "applicants are free and indeed encouraged to demonstrate to the Commission the applicability and significance of considerations other than those set forth in the guides."

Other changes in the criteria were incorpo-

rated in an attempt to clarify the intent of the Commission on specific subjects and to cover additional points that were raised by the comments. Thus some further clarification is now included on the following issues: (1) multiple reactors at one site, (2) dependence upon isolation versus engineered safety features, (3) applicability to mobile reactors, (4) effect of population growth, and (5) population-center-distance concept, as follows:

1. The siting criteria for multiple reactors at one site may be determined on the basis of the evaluation of individual reactors if the reactors are independent to the extent that an accident in one does not result in a simultaneous accident or disruption of operation of adjacent reactors.

2. The revised guides still reflect the Commission's emphasis on isolation, although the phraseology has been modified to indicate a potential willingness to accept engineered safety features on proved plants in lieu of isolation.

3. Although the general safety concepts that are embodied in these criteria may be expected to be applicable to all power reactors, these guides were not meant to be directly applicable to other than stationary plants and have been restricted accordingly.

4. No guidance is given on the implications of future population growth in the vicinity of a reactor, although it may be implied that each nuclear installation should acquire sufficient experience with engineered safeguards to make it possible to rely on such factors rather than isolation as the surrounding population increases.

5. The population-center-distance concept, although not entirely satisfactory, reflects an accurate guide to current siting practices. It was said to be retained in lieu of any better

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technique of both (a) limiting the total population exposure and (b) providing protection against excessive exposure doses to people in large centers where protective measures may not be feasible. The latter reason, however, reflects a lack of confidence in containment systems, inasmuch as the man-roentgen equivalent man (rem) concept suggested both here⁶ and by the Advisory Committee on Reactor Safeguards⁷ is a direct means of specifying population exposure.

Stipulations of Current Criteria

The published site-criteria document does not differ substantially in principle from the previous draft. The criteria in effect define three distances from the reactor which must meet certain conditions. These distances are (1) the exclusion area, (2) the low-population zone, and (3) the population-center distance. The population-center distance is defined as the distance from the reactor to the nearest boundary of a densely populated area containing more than about 25,000 residents and is somewhat arbitrarily taken as $1\frac{1}{2}$ of the low-population zone. On the other hand, both the low-population zone and the exclusion area are related to exposures, which are calculated at the outer perimeter of those areas following the maximum credible accident. The exclusion area is the area immediately surrounding the reactor which is under the control of the reactor operator and which is of such size that an individual located at the boundary for 2 hr immediately following the maximum credible accident would not receive a total radiation dose in excess of 25 rem to the whole body or 300 rem to the thyroid. The low-population zone is the area immediately surrounding the exclusion area. The low-population zone may contain residents, but the total number and density must be such that there is a reasonable probability that appropriate measures could be taken in their behalf in the event of a serious accident. This zone is of such size that an individual located at any point on its outer boundary during the entire time of passage of the activity released from the maximum credible accident would not receive a total radiation dose in excess of 25 rem to the whole body or 300 rem to the thyroid.

The site criteria also list the following factors that are to be taken into consideration

when evaluating sites (and hence also when calculating exposures at the specified distances):

1. Characteristics of the reactor, which include the utilization of the reactor, design of the facility, features affecting activity release, and engineering safety features
2. Population density and use characteristics of the site environs, which include the exclusion area, low-population zone, and the population-center distance
3. Physical characteristics of the site, which include seismology, meteorology, geology, and hydrology

Calculational Procedures

The example of the calculational procedures, which was included in the second draft as an appendix, has been deleted from the criteria and has been expanded and incorporated into a separate report.⁵ The report recognizes that there will be significant differences between different reactors and different containment systems. The analytical example that was presented referred specifically to the maximum credible accident, which was defined for pressurized- and cooling-water reactors. It was noted that other types of reactors would have "to postulate...corresponding maximum credible accidents...by similar analyses." The fundamental assumptions on which the distances are calculated, with estimates of the degree of conservatism represented in each case, are quoted⁸ in the following:

1. 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.⁽⁹⁾ The probability is still lower for an accident in which significant amounts of fission products are released into the confined primary coolant system and a great deal lower for accidents which would release significant quantities of radioactivity from the primary system into the reactor building.
2. It is assumed that the reactor is a pressurized water type for which the maximum credible accident will release into the reactor building 100 percent of the noble gases, 50 percent of the halogens and 1 percent of the solids in the fission product inventory. Such a release represents approximately 15 percent of the gross fission product activity.
3. Fifty percent of the iodines in the containment vessel is assumed to remain available for release to the atmosphere. The remaining fifty percent of

the iodines is assumed to absorb onto internal surfaces of the reactor building or adhere to internal components. Rather than the assumed reduction factor of two, it is estimated that removal of airborne iodines by various physical phenomena such as adsorption, adherence and settling could give an effect of 3-10 reduction in the final result. Credit has not been taken for the effects of washdown or filtering from protective safeguards such as cooling sprays and internal air recirculating systems. Washdown features and filtering networks could provide additional reduction factors of 10-1000.

4. The release of available (airborne) radioactivity from the reactor building to the environment is assumed to occur at a constant leakage rate of 0.1 percent per day. The leakage and pressure conditions are assumed to persist throughout the effective course of the accident, which for practical purposes, would be until the iodine activity becomes insignificant. The maximum pressure within the reactor building and the leakage rate would actually 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, it is estimated that the final off-site doses calculated may be too high by factors of 5-10.

5. Atmospheric dispersion of material from the reactor building is assumed to occur according to the well-known relationship developed by O. G. Sutton^[9] involving meteorological factors of wind velocity, atmospheric stability, and diffusion parameters. Application of this treatment to reactor hazards analysis was discussed in WASH-740^[10] and AECU-3066.^[11] Recently a simplified method of dispersion calculation has been proposed, by Pasquill^[12] and Meade,^[13] which reflects recent dispersion field trials, as well as current dispersion theories. This method gives the same numerical results as the Sutton method in the present application to a distance of about seven miles. Beyond this distance, the new method predicts somewhat greater concentrations.

6. The assumption that no shift in wind direction does not occur for the duration of the leakage of the fission products from the containment barrier. If leakage from the containment barrier is assumed to occur over a significant time period (in the order of days), a reduction factor of 2-50 could result from shifts in wind directions. Wind meandering from any one centerline direction might also result in a reduction factor of approximately 3.

7. Atmospheric dispersion is assumed to occur under inversion type weather conditions. For weather conditions which exist for 75 percent or so of the time at most sites, the atmospheric dispersion conditions could be more favorable, by factors of 5-1000.^[14]

8. Cloud depletion as ground deposition (particulate fallout) is not assumed during cloud travel.

Such deposition during cloud travel could reduce the low population zone distance by factors of 2-5.

9. In calculating the direct gamma dose, credit is not taken for shielding by the containment structure and applicable reactor shielding or topography. In some cases it is recognized that such shielding could reduce the direct gamma dose by a factor of 2-1000.

10. Decay of fission products is assumed while they are confined to the containment building but is not assumed during their transit to the receptor point. The decay enroute is not significant for the conditions of release considered here but would lower the calculated doses slightly if included.

11. In determining the whole body direct gamma dose, only the external gamma dose due to the fission products contained in the reactor building was considered significant for the assumed conditions. The whole body direct gamma dose due to the cloud passage for the assumed conditions would contribute on the order of 1-10 percent of the total whole body direct gamma dose at the exclusion and low population zone distances.^[15]

The report⁵ explains that:

...even if the postulated maximum credible accident should occur and if the external containment shell remained intact, the resulting exposure doses would probably be many times lower than those calculated by the indicated method.

On the other hand [Ref. 5 continues], there are potential, conceivable conditions which would result in larger fission product releases than those assumed to be released in the maximum credible accident, and the consequences could be more hazardous. Other potentially more hazardous factors than those represented by the example calculation include the following conditions.

1. Total radioactivity releases to the containment vessel could theoretically be up to six times as large as those assumed. Release of long-lived fission products to the containment vessel could theoretically be up to 99 times as large as that assumed. These greater releases would affect doses to the lung, bone, and total body.

2. For some sites, the atmospheric diffusion conditions for a small proportion of time could be worse than those assumed in these calculations. Such diffusion conditions could result in an increase in the inhalation doses.

3. If the external containment structure should be rendered completely ineffective at the outset of the accident, the consequences of the "maximum credible" accident would be increased many orders of magnitude. In such a case, the dose from the cloud and ground contamination could become significant in determining the external dose.

With the above frame of reference, the report then defines, in detail, the analytical techniques and procedures common to most reactor hazards reports for calculating external and internal exposures from a fission-product cloud. Data are presented for meteorological conditions, as given below, which are indicative of the slow dispersion conditions expected to apply between 15 and 25% of the time in most areas of the United States.

Wind speed, m	1 m/sec
Horizontal virtual diffusion coefficient, C_y	$0.40 \pi^{1/2}$
Vertical virtual diffusion coefficient, C_z	$0.67 \pi^{1/2}$
Stability parameter, π	0.5 (Dimensionless)

The results of calculations employing a containment-structure leakage rate of 0.1%/day were used to determine reference values of distances for water-cooled and -moderated reactors of various power levels. The reference values thus obtained are listed in Table I-1.

The estimated radii for power reactors are graphically represented in [Figs. I-1 and I-2]. For the exclusion distance, doses from both direct gamma radiation from the reactor building and from iodine in the cloud escaping from the reactor building were calculated, and the distance established on the basis of the effect requiring the greater isolation. [Figure I-1] shows the controlling dose for various power levels.

Table I-1 SITE-CRITERIA RADII FOR REACTORS OF VARIOUS POWER LEVELS

Thermal power level, Mw	Exclusion-area radius, miles	Low-population-zone distance, miles	Population-center distance, miles
1500	0.88	13.3	17.7
1200	0.77	11.5	15.3
1000	0.67	10.3	13.7
900	0.63	9.4	12.5
800	0.58	8.6	11.5
700	0.53	8.2	10.9
600	0.48	7.2	9.6
500	0.43	6.5	8.7
400	0.37	5.4	7.2
300	0.31	4.5	6.0
200	0.29	3.4	4.5
100	0.25	2.2	2.9
50	0.21	1.4	1.9
10	0.12	0.5	0.7

Fig. I-1 Exclusion area-radius determination.

Under the conditions assumed, the doses resulting from the inhalation of the isotopes of iodine are controlling for the low population zone distance and population center distance. However, it is possible that such may not always be the case and this should be checked for each case under consideration. The low population zone distance results from integrating the effects of iodine 131 through 135. The population center distance equals the low population zone distance increased by a factor of one-third. [Figure I-2] shows the relationship between the inhalation and direct dose for various power levels.

The Division of Technical Information document⁵ concludes with a comparison of the site distances that were obtained by the above analytical method with those which have been authorized for operating reactors or reactors authorized for construction. These values are given in Table I-2.

Conclusions

One may justly conclude, as the Commission has done, that although it is generally recognized that insufficient experience with reactors has been accumulated to permit the writing of definitive standards that would provide a quantitative correlation of all factors significant to the question of acceptability of reactor sites, it is possible to provide more guidance than currently exists as to the fac-

tors considered by the Commission in evaluating reactor sites. The proposed guide ... is intended as an interim measure until the state of the art allows more definitive standards to be developed. The guide is a starting point that gives the nuclear community an understanding of the basis on which the Commission will review proposed sites while providing flexibility and freedom for any applicant to demonstrate the applicability and significance of considerations other than those set forth in the guides. The proposed guides do not represent a different approach to reactor siting than has been used to date but rather they represent an attempt to articulate those practices. Application of these criteria will result in reactor sites in general agreement with those approved to date but sufficient flexibility has been provided to allow for considerations that might lead to locations that vary from current practice. [From Ref. 5.]

It is of interest to note that both the conservative and nonconservative factors that are involved in the determination of exposures at the specified distances are tabulated in Ref. 5. Excluding the assumed atmospheric dispersion (which appears on both lists), the conservatism in the calculation of inhalation exposure may range from 10^3 to 10^6 . Against these are ranged three "potential conceivable conditions" that result in higher exposures. These conditions are (1) greater activity release, (2) poorer atmospheric diffusion, and (3) simultaneous failure of the containment structure.

Fig. 1-2 Population-radius determination.

Whether or not these three factors are conceivable, as suggested, is a question of semantics. Thus it is theoretically possible for the total activity release to be up to six times as large as that assumed (since the release assumed was approximately 15% of the total). However, many responsible technical people

Table 1-2 CALCULATED AND ACTUAL SITE DISTANCES FOR SELECTED REACTORS

Reactor	Thermal power level, Mw	Exclusion-area radius, miles		Low-population-zone distance (calculated), miles	Population-center distance, miles	
		Calculated	Actual		Calculated	Actual
Dresden	630	0.50	0.50	7.4	9.9	14.0
Consolidated Edison	585	0.48	0.30	7.0	9.4	17.0
Yankee	485	0.42	0.50	6.3	8.4	21.0
PRDC	300	0.31	0.75	4.5	6.1	7.5
PWR	270	0.51	0.40	4.1	5.6	7.5
Consumers	240	0.30	0.50	3.9	5.2	135.0
Hallam	240	0.30	0.25	3.9	5.2	17.0
Pathfinder	203	0.29	0.50	3.4	4.6	3.5
PG&E	202	0.29	0.25	3.4	4.6	3.0
Philadelphia Electric	115	0.26	0.57	2.4	3.2	21.0
NASA	60	0.22	0.50	1.6	2.1	3.0
CVTR	60	0.22	0.50	1.6	2.1	25.0
Elk River	58	0.22	0.23	1.5	2.0	20.0
SWR	50	0.21	0.40	1.4	1.9	15.0
Piqua	48	0.21	0.14	1.4	1.8	27.0

would conclude that 100% release is not conceivable in view of (1) the fuel-meltdown experiments that have been performed, (2) the limit on the attainable temperature, even in a core meltdown, and (3) the known chemical reaction behavior of the fission-product elements. On the other hand, the second nonconservative factor—poorer atmospheric diffusion—is not only conceivable but has a 25% probability of occurrence, which must be weighed against the other conservative factors that are involved in the calculation.

The point of greatest concern, however, involves "conceivable" accidents that could render ineffective the containment vessel or, for that matter, any other engineered safety features. This question lies at the center of site-evaluation policy today. Present technology is adequate to provide containment or safety features, or both, for nuclear facilities at only nominal increases in cost which would permit the location of such facilities in population centers and still meet the existing requirements for exclusion area and low-population zone. This possibility is precluded, however, in the present site criteria by the specified population-center distance, which is included as a direct result of the lack of confidence in the integrity of the existing containment systems in the time of need and which led the AEC to consider the consequences of such an eventuality. The reviewer anticipates that, as confidence in the integrity of containment systems (as well as other engineered safety features) is developed through experience and as the need for nuclear power increases, the present site criteria, which are based on distance, will give way to criteria that will be based on total population exposure. Such site criteria would still employ the exclusion-area concept as a means for limiting the maximum exposure to any one individual, and the low-population zone and population-center distances would evolve into a total-population-exposure requirement. Then the task of the AEC will be to establish criteria for containment systems and engineered safety features, and such criteria are possible.

It will be many years, however, before suf-

ficient experience has been developed to permit this reorientation of siting policy. In the meantime the existing site criteria provide a workable and generally consistent basis for siting reactors. The criteria codify existing practice and permit the economic assessment of nuclear power in the light of a clearly defined siting policy.
(Wm. B. Cottrell)

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2.2 Comments on Proposed Reactor Site Criteria

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[*Nucl. Safety*, 3(2): 73-76 (December 1961)]

Proposed criteria for use as guides in the selection of reactor sites were published in the *Federal Register* on Feb. 11, 1961, and comments or suggestions addressed to the Division of Licensing and Regulation of the AEC were invited from interested parties. The criteria and previous siting developments were reviewed in the June 1961 issue of *Nuclear Safety*.^{19,20} Even if the proposed criteria were adopted in anything like the present form, the judgment of the designer, the AEC regulatory staff, and the ACRS remains vital to the selection of a site, since, as acknowledged by AEC, site criteria cannot yet be stated precisely.

Comments were received from 30 sources, including ACRS, the Atomic Industrial Forum, and foreign, as well as American, groups. The Committee on Reactor Safety of the Atomic Industrial Forum on Mar. 17, 1961, discussed the criteria in a special Forum seminar participated in by 17 of its members.²¹ In addition to comments, the Forum submitted,²² as did Atomics International,²³ a proposed complete revision of the criteria. The selected quotations appearing in the discussion that follows were taken from many of the comments in an attempt to indicate both their range and depth. Although it is intended that the original meaning of the comments be maintained, all are, of course, presented here out of context, and persons interested in greater detail should refer to the original comments. It is anticipated that the Division of Licensing and Regulation will embody many of the comments and suggestions into a revision of the criteria which will then be issued in the *Federal Register*.

General Comments

For the most part the groups who commented welcomed the criteria, since they saw the need for guide lines for the industry. There was, however, much fear expressed that the criteria were too restrictive and that, once they were set forth in the *Federal Register* as regulations

and were fixed in the minds of the public and authorities, it would be difficult to make site decisions counter to the criteria even though such decisions might be warranted by the particular designs. The use of the calculated example (in Appendix A of the proposed criteria) for a serious accident of a hypothetical reactor was a cause of concern. It was felt that too much emphasis was placed on distances to populated centers, to the detriment of the engineering safeguards that can be developed. A sampling of the general comments follows:

... use of the term "guides" may lead to some ambiguity as to their intent and purpose ... the proposed guides had been identified by AEC as ... (part) of the *Code of Federal Regulations* ...

There was an obvious difference of opinion among seminar participants as to the intended purpose of Appendix A. ...

... the guides may, through inadvertent phraseology, have a possible adverse impact on public opinion even though their intent might be clearly understood by reactor licensees and others in the atomic industry.^[21]

We believe that the guides outlined in the proposed criteria are most desirable for preliminary evaluation of potential reactor sites. ...

An effort should be made to establish the trade-offs which can acceptably be effected between engineering safeguards incorporated in the reactor system and the various distances specified in the present guides.^[24]

To my knowledge, there is no definite requirement by the Commission which would prevent an industrial park or suburban housing development from springing up around a reactor, once it was licensed, and effectively abolishing the "low population zone" so carefully calculated as a requirement for issuing the license.^[25]

It is apparent throughout the proposed guide that it was written for a specific class of reactors and a specific type of containment. This is unfortunate.^[26]

Objective (b) of the proposed guides is not clear because the phrases "not normally considered

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credible" and "the number of people killed should not be catastrophic" are subject to a considerable range of subjective interpretation. In addition, the latter could provoke public alarm without need.

It is recommended, therefore, that the basic objective in the establishment and use of the proposed guides be stated as:

"Serious injury to individuals offsite should be avoided and the exposure of large numbers of people in terms of total integrated population dose should be low, if any unlikely, but still credible, accident should occur."²¹

... the document recognizes tacitly that the ACRS and the AEC as a whole are not as yet completely willing to trade distance for engineering design. This is a reasonable position at this stage in the development of the atomic energy industry and in view of the projected future population growth.

The intent and desire of the ACRS is that this be a very flexible document and that deviations from this document should be allowed in cases where an applicant can show that his design or his siting of the reactor leads to a safe situation. In fact, the ACRS would like to encourage applicants to come in with deviations in those cases in which they believe that a valid reason for deviation exists.²²

Specific Comments on Guide

Concern was expressed about some special cases in which application of the proposed concept of a "population center distance" would practically exclude power reactors from areas where most authorities feel they will be utilized. Japan feels that its regulations are likely to evolve from the U. S. regulations, and, in Japan, it would almost be impossible to find power-reactor sites based on the proposed regulations because of the high population density. It was also pointed out by several sources that the regulations created an awkward situation for mobile reactor power plants, such as that of the N.S. *Savannah*. The situation with respect to process-steam reactors would also be clouded. Some of the comments on these special cases follow:

The proposed guides on reactor site criteria published by the United States Atomic Energy Commission on February 11 this year [have] brought about a great deal of repercussion among the related quarters of the Japanese industry. We consider that it is now a sort of world wide common sense that in the field of nuclear energy even domestic legislations cannot be made without influencing or being influenced by the situation abroad.²³

Should these figures of exclusion area, low population zone, and population center distance as appearing in the Appendix A be adopted for granted, it would eventually make it practically impossible to find a site in Japan for installing any U. S. type reactor.²⁴

I would like to note that the Commission's efforts to develop industrial process steam reactors also appear to raise questions about the consistency of AEC's proposed guides on site criteria. As you know, it is not economically practical to transmit steam for long distances. In most cases, it is necessary that the industrial plants using process steam be located closely around the steam-producing plant. The Commission's distance-from-population criteria for reactors seem to us incompatible with the Commission's expenditures of time and money for the development of process steam reactors....

There seems to be another inconsistency, as far as remoteness from population is concerned, in the Commission's attitude in respect to nuclear-powered ships.... I recognize that the [N.S.] *Savannah* has been designed and built with special features to enhance the inherent safety of the power plant...²⁵

Comments on Multiple Reactor Sites

The question of how to apply the criteria to multiple reactors was introduced. Some of the comments were:

is the power level to be used for determining distances for population the total power level of all nuclear units on site or only the power level of the largest unit?²⁶

Where more than one reactor is located at a particular site, the thermal rating used in hazards calculations pertaining to the total installation should be somewhere between the thermal rating of the largest single reactor and the sum of all reactors present. If there is negligible possibility that an accident in one reactor could cause an accident in another at the same site, then the thermal rating of the largest single reactor should be the value used in hazards calculations.²⁷

Comments on Evaluation Factors

There were several suggestions that the emphasis on distance to population centers as a factor used in evaluating reactor sites should be replaced by an emphasis on the safety of the engineering design. Geological and hydrological factors were discussed in relation to the precautions to be taken to prevent radioactive effluents from running into rivers or reaching

underground water tables. Criticism and questions pertaining to the maximum emergency radiation-exposure dose for the population were included among the remarks. Some of the comments were:

We view with concern the site evaluation factors. . . . This approach appears to minimize the possible contributions to safety by advances in design and to maximize the value of isolation. We believe this approach unrealistically undervalues scientific and engineering skill, both present and future.⁽³¹⁾

One fundamental point is that the contribution which site considerations can make to the safety of the public is relatively small, when compared to the contribution made by engineering barriers to the release of fission products. . . .

We regard the proposed population center distance factor as technically unjustified and recommend the development of a rating system which factors in meteorology and other environmental factors, in addition to distance. Such an approach provides a much greater degree of assurance that population center distance will make a significant contribution to public safety.⁽³²⁾

It seems very unlikely that future inland sites can be found which have water supplies which are not used at some downstream point. . . . We would suggest "The plant design must provide reasonable assurance that radioactive liquid effluents cannot accidentally contaminate usable water supplies and prevent use of the water for long time periods. The provisions to prevent accidental releases as well as the importance of the potentially contaminated water must be considered." . . .

In our opinion, some method of evaluating the risk of exposing large numbers of persons should be developed taking into account the integrated man-rem exposure potential based on the proposed plant including its design features, the meteorology and the size, distance, and direction of areas of substantial population.⁽³³⁾

We do not question the 25-rem whole-body exposure, but we do believe that the 300-rem thyroid exposure is too high. The highest value iodine exposure equivalent that we could justify at the present time would be 150 rem. . . . We suspect that the higher thyroid values were included in the proposed criteria on the basis that exposed persons may be treated in such a fashion as to reduce their thyroid exposure by a factor of 2. However, the use of the higher value would be justified for a particular reactor only if that installation were prepared to, and were capable of, rounding up all the persons

which had been evacuated from the low-population zone and getting them to submit to the required treatment. This does not appear to be either desirable or practical.⁽³⁴⁾

The most important single new development which is embodied in this document is the fact that it contained a set of dosage limits which can be used in the event of a very unlikely reactor accident. . . .

The site criteria, as published, do not completely define or take care of the problem of either genetic or somatic damage. The problem is recognized and stated in the document, but answers to this problem do not as yet exist. . . . The Committee [ACRS] believes that the so-called "man-rem" dose has much to offer as a safeguard for the general protection of the population. We went so far, in December of 1960, as to suggest to the Commission that a dose limit of 1×10^6 man-rem per accident might be reasonable.⁽³⁵⁾

Comments on Appendix A

The example in Appendix A, which gave calculations for possible reactor siting distances, was the most controversial portion of the proposed regulation. Some of the comments received with respect to the example follow:

. . . the AEC's Division of Licensing and Regulation has clearly demonstrated a tendency on the part of AEC to take the most conservative of alternative interpretations permitted by any rule. Hence, unless there is an unequivocal intent on the part of the AEC not to permit variance from the assumptions used in Appendix A, it should not be included as an integral part of the guides.⁽³⁶⁾

. . . we also believe that the example at the end should be treated simply as an example and, perhaps, should be either deleted or augmented by the addition of several other examples.⁽³⁷⁾

The proposed criterion tabulates conversion factors which were used to determine the dose received from breathing a specified quantity of the various iodine isotopes. We believe that the actual conversion factors are appreciably higher and suspect that the lower values in the guide were calculated from an earlier inaccurate value of the uptake of iodine by the thyroid as well as a lower breathing range.⁽³⁸⁾

We do not agree with the general applicability or the technical validity of the proposed Appendix A and urge that the Appendix be deleted.⁽³⁹⁾

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2.3 Site Criteria for Reactors With Multiple Containment^{*}

W. K. Ergen[†]

[*Nucl. Safety*, 4(4): 8-14 (June 1963)]

Design criteria used in evaluating a power-reactor site are usually dominated by concern about radioactive pollution of the atmosphere in the case of the "maximum credible accident," which is assumed to involve extensive meltdown or vaporization of the core. In the United States, criteria determined on this basis are set forth in the *Code of Federal Regulations*¹ and supplemented by an AEC report² that describes the calculation of distance factors. These docu-

ments define an exclusion area in which residence will normally be prohibited; a low-population zone, the inhabitants of which are subject to protective measures such as evacuation; and a population-center distance that is the distance to the nearest boundary of a densely populated center. The following minimum values for the radii of the exclusion area and the low-population zone and for the population-center distance can be computed, based on assumptions

^{*}The views expressed in this article do not necessarily represent those of the U.S. Atomic Energy Commission or its appropriate office or division.

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stated explicitly in the AEC document:^a

Radius of exclusion area = $11 \times (\text{reactor power})^{\frac{1}{2}}$
 Radius of low-population zone = $161 \times (\text{reactor power})^{\frac{1}{2}}$
 Population-center distance = $215 \times (\text{reactor power})^{\frac{1}{2}}$

The distances are in meters, and the reactor power is in thermal megawatts. These relations are plotted in the "distance graphs" of Figs. 1 and 2 of Ref. 2. It is clear in the documents referenced^{1,2} that many factors enter into the determination of the suitability of a site and that "applicants are free and indeed encouraged to demonstrate to the Commission the applicability and significance of considerations other than those set forth in the guides." The power level for which a given reactor is suitable on a given site has to be determined on an individual basis. This power level may deviate from the power determined by the above relations by a "safeguard factor." This factor could be smaller than 1, if the reactor's engineered safeguards were less than those assumed in Ref. 2, and it could be greater than 1 if the engineered safeguards of the individual reactor exceeded those assumed in the AEC document. For the purpose of discussion of a specific engineered safeguard, it is, in principle, possible to talk about the safeguard factor associated with this specific safeguard feature. This permits comparison between a reactor incorporating the feature and a reactor without it. In practice, the situation is more complicated, but the concept of the safeguard factor associated with a specific safeguard is useful.

^aFor instance, the AEC report² assumes containment of the reactor that permits maximum leakage after the maximum credible accident of 0.1% per day of the reactor's rare-gas inventory and 0.025% per day of the halogen inventory. Other radioisotopes are assumed to be held up sufficiently so that their contribution to the hazard is of no concern. The gamma dose from fission fragments dispersed in the containment vessel by the postulated accident is disregarded in this article because this dose can be eliminated by shielding. An individual located at the boundary of the exclusion area is assumed to be exposed to the consequences of the accident for 2 hr immediately following the onset of the postulated fission-product release (thereafter the protective measures would have taken effect), whereas an individual located at the outer boundary of the low-population zone is assumed to be exposed to the postulated fission-product release during the entire period of its passage. The population-center distance is 1.33 times the low-population-zone radius.

Multiple containment is one of the devices used to drastically reduce the release of iodine, and it can also introduce a delay in the release of rare gases. If it operates as intended, it makes a given site suitable for a higher reactor power than the "standard" containment vessel on which the present U. S. site criteria are based. The general principles involved in multiple containment are discussed here. Specific examples are given with respect to the following reactors: the Consolidated Edison Thorium Reactor; the N.S. Savannah reactor; the reactor recently proposed by the City of Los Angeles, Department of Water and Power; the reactor proposed by the Consolidated Edison Company of New York; and the very large reactors proposed for desalting sea water.

Iodine and Noble Gas Exposures

In order to place reactors in regions having a greater population than would be compatible with the present criteria, consideration has been given to drastically reducing the possibility of iodine release. The potential air pollution would then be dominated by the rare gases, at least under the assumptions of Ref. 2 with respect to radioisotope release. The rare gases constitute a submersion-exposure hazard; i.e., an individual immersed in the cloud of radioactive rare gases would receive radiation from the cloud's atoms.

Release of the radioactivity is assumed to occur at ground level. The radioactive cloud is narrow at the source. As the cloud reaches greater and greater distances from the source, it spreads more and more. For any given distance from the source, there is a point of maximum concentration and, corresponding to the spreading, the maximum concentration decreases with increasing distance. It is mathematically simple to compute the dose to an individual as if he were immersed in a semi-infinite cloud of constant concentration. This concentration is assumed to be the maximum concentration at the actual distance of the individual from the source. The semi-infinite cloud concept would apply if the individual were standing on a flat surface, with the cloud above him and no radioactivity below him, and if the ground scattered radiation to the same extent as the

atmosphere.* This simple procedure overestimates the exposure because the conservative assumptions regarding the meteorological conditions at the time of the accident imply relatively large centerline concentrations, which are obtained only in very narrow clouds.

If the simple procedure is used, the dose rate, in rem per second, is

$$D_r = 0.26 E \chi$$

where E is the effective energy in millions of electron volts per disintegration and χ is the concentration in curies per cubic meter.³ (In order to make a consistent presentation, the units and the numerical factor in Burnett's¹ equation were changed to agree with those of Ref. 4.) For any given isotope the concentration χ , is given by

$$\chi_i = 0.865 \times 10^4 P_0 \gamma_i F \lambda_1 B e^{-\lambda_{r,i} t}$$

where P_0 = reactor power (Mw)

γ_i = fission yield of the isotope

F = fraction of the isotope inventory available for leakage from the reactor building (assumed to be 1 for rare gases)

λ_1 = fractional rate of leakage from the reactor building to the atmosphere (sec^{-1})

$\lambda_{r,i}$ = radiological decay constant of the isotope (sec^{-1})

t = time since shutdown of the reactor (sec)

B = distance-dependent dilution factor giving the concentration in curies per cubic meter caused by the leakage of 1 curie/sec from the reactor building

Infinite reactor operation prior to the shutdown is conservatively assumed, and $\lambda_{r,i}$ is assumed to be $\gg \lambda_1$, as it is, indeed, for the isotopes of interest. (The expression $\lambda_{r,i} \gg \lambda_1$ means that decay is more important than reduction of the inventory by leakage from the containment.)

Thus the dose rate caused by the rare gases is

$$D_r = 7.0 \times 10^{-12} P_0 \lambda_1 B \sum S_0 e^{-\lambda_{r,i} t}$$

where

$$S_0 = 3.2 \times 10^{16} E \gamma$$

is the initial source strength of the isotope in question in millions of electron volts per second per megawatt. The S_0 values are tabulated in Ref. 2 (Table IV, page 27, column 4), the work of Blomeke and Todd.⁵

The dose rate must be integrated over the time during which an individual located at the outer boundary of the exclusion area or of the low-population zone is exposed to the radioactive cloud. If no significant time delay occurs between the shutdown of the reactor and the moment at which the cloud reaches these boundaries, the integral over the time-dependent exponential becomes $(1 - e^{-\lambda_{r,i} t})/\lambda_{r,i}$ for the postulated 2-hr exposure at the exclusion-area boundary; for long exposure at the boundary of the low-population zone, the integral is essentially $1/\lambda_{r,i}$.

If a delay in fission-fragment release from the reactor building in excess of 2 hr could be assured, the individuals in the "standard" low-population zone could presumably be evacuated and the inner boundary of the low-population zone could be moved inward to the point where the external gamma-radiation dose from fission products contained in the reactor building would reach 25 rem during the first 2 hr after the incident. Since the reactor building could be shielded to any desired extent, the inner boundary of the low-population zone (or outer boundary of the exclusion area) would lose much of its meaning. For the determination of the outer boundary of the low-population zone, the integrated dose rate becomes $e^{-\lambda_{r,i} \tau}/\lambda_{r,i}$ if no radioactivity is released prior to time τ (in seconds) following the accident and if after this time the fractional rate of leakage λ_1 applies. More sophisticated delay models have been investigated

*The semi-infinite cloud is used here because of a corresponding assumption in the iodine-dose calculations of Ref. 2. For a given release of radioactive material, the concentration in a semi-infinite cloud is twice as high as the concentration in an infinite cloud; however, an individual immersed in a semi-infinite cloud receives radiation only from above, whereas an individual in an infinite cloud receives radiation from above and below. Thus the semi-infinite cloud and the infinite cloud actually give the same dose.

by Anderson.⁴

Numerical evaluation yields, for the case of no delay in the release,

$$D_1 = 1.51 \times 10^8 P_0 \lambda_1 B$$

for the exclusion-area boundary. Two-thirds of this dose is contributed by Kr⁸⁸. For the outer boundary of the low-population zone,

$$D_1 = 12.56 \times 10^8 P_0 \lambda_1 B$$

62% of this is contributed by Xe¹³³ and 20% by Kr⁸⁸.

For iodine, it follows, from Ref. 2, Eqs. 1 and 9, that the dose to the thyroid is given by

$$D_{\infty} = P_0 B R \sum Q_i (D_{\infty}/A_i)$$

where R is the breathing rate, which is assumed to be 3.47×10^{-4} m³/sec for the exclusion zone and 2.32×10^{-4} m³/sec for the low-population zone. The expressions Q_i and (D_{∞}/A_i) are defined in Ref. 2, and their values are tabulated in Tables II and III, respectively, of Ref. 2. The values are based on the assumption that one-fourth of the iodine inventory is available for leakage from the reactor building and that the leak rate from this building is 0.1% per day. The subscript i is used here to distinguish the various iodine isotopes.

The sum $\sum Q_i (D_{\infty}/A_i)$ is 1.4×10^6 rads/Mw for the exclusion zone and 116×10^6 rads/Mw for the low-population zone. Thus the dose to the thyroid is

$$D_{\infty} = 4.19 \times 10^{10} P_0 \lambda_1 B$$

for the boundary of the exclusion zone and

$$D_{\infty} = 232 \times 10^{10} P_0 \lambda_1 B$$

for the outer boundary of the low-population zone.

In the reactor site criteria, a 300-rem dose to the thyroid is used in the same context as a whole-body dose of 25 rem. It is thus of interest to compare ²⁵/₂₅₀, or ¹/₁₂, of the thyroid dose with the whole-body dose. From the above it may be seen that

$$\frac{1}{12} D_{\infty}/D_1 = 22.9$$

for the exclusion radius and

$$\frac{1}{12} D_{\infty}/D_1 = 153$$

for the outer boundary of the low-population zone. These ratios are independent of the power level, the assumed leak rate, and the dilution factor B . A given location that the distance graphs regard as compatible with a power level P_0 would be compatible with a power level that exceeded P_0 by a safeguard factor. This factor is 22.9 if the exclusion radius is limiting and 153 if the radius of the low-population zone is the determining item. Similar calculations have been made by Binford.¹

As previously mentioned, the factor could be still higher, because the assumption of a semi-infinite cloud is unduly conservative. Finite-cloud external gamma doses can be found from the nomograms in Report AECU-3066, Meteorology and Atomic Energy,⁴ Chap. 8, Sec. 3. These nomograms are based on the assumption that all fission fragments, not only the rare gases, are released. Anderson and his co-workers⁵ have used these nomograms to compute the ratio of the finite-cloud doses to the doses resulting from an infinite cloud under the assumed meteorological conditions of Ref. 2. At 300 m from the source, the finite-cloud dose is only one-tenth of that resulting from an infinite cloud, and closer to the source the ratio of the doses from finite and infinite clouds is still smaller.

If holdup of the fission fragments in the containment vessel is provided, the exposure from the rare gases is also reduced by radioactive decay. After a few days following the postulated release, this decay is controlled by the 5.27-day Xe¹³³, until, finally, longer-lived rare gases become dominant.

In summary, a given site would become compatible with a much higher power level than that given by the distance graphs if the iodine could be eliminated. The large factors by which the power level might thus be increased are contingent on the confidence that can be placed on the containment of iodine and other fission fragments, such as particulates.

Multiple Containment

As discussed in the preceding subsection, the potential consequences of an accident can be greatly reduced by (1) drastically reducing the radiiodine content of the effluent from the containment vessel and (2) providing a time delay

between the accident and the release of any effluent to allow the rare gases to decay. One of the most effective and most frequently proposed means to accomplish these two objectives is multiple containment. In its simplest form the multiple-containment scheme incorporates two barriers around the core. This provides three zones: (1) the "inner" zone inside the inner barrier; this zone contains the core and would be heavily contaminated by the postulated accident; (2) the "intermediate" zone between the two barriers; and (3) the outside atmosphere surrounding the outer barrier.

The intermediate zone is kept at "negative" pressure, i.e., at a pressure below the pressure in the inner zone and below the pressure of the outside atmosphere. The barriers are made as leaktight as reasonably possible, but there will be some leakage. This leakage will be into the intermediate, negative-pressure zone, and theoretically there should be no uncontrolled leakage from the contaminated inner zone to the outside atmosphere.

In order to maintain the intermediate zone at the negative pressure in spite of the inleakage, some gas has to be pumped from this space. This gas will be contaminated because it will contain whatever leakage occurs from the inner space. Two possible means have been proposed for disposal of the contaminated gas:

1. Release to the atmosphere after filtration of the iodine. The inner barrier protects the filters from high temperature, and the filters only have to handle the relatively small amount of fission fragments leaking through the inner barrier. Under these circumstances the filtration can readily be performed with high efficiency, and the iodine content can usually be reduced to the point where the rare gases become the dominant hazard. The filters can, of course, also be designed to remove, with high efficiency, other fission fragments, such as cesium and strontium, making the assumption of negligible release of these isotopes more realistic. Retention of the rare gases on refrigerated charcoal or other filtering agents would be possible, but so far this has not actually been done.

The release of the rare gases from an elevated stack, with possibly some additional dilution and some delay on the path from the reactor to the top of the stack, offers some advantage over and above that resulting from the fact that the rare gases are less harmful than the radio-

iodine. The inherent limitation of this scheme is that the remaining exposure from the rare gases must be within limits acceptable for the case under consideration.

2. Storage of the contaminated gas on the site. This could be accomplished, for instance, by pumping the gas back into the inner zone, since this is a large volume and readily available. There would be some leakage from the outside atmosphere into the intermediate zone, and this leakage would be pumped into the storage volume. This would increase the pressure in the storage volume, and, since there would be an upper limit on the pressure for which the storage volume was designed, there would be an inherent limitation on the duration of storage. When the pressure reached the design limit, some relief to the atmosphere would be necessary, unless additional storage space was provided by that time. Even if the radioactive effluent could not be held up indefinitely, the delay, coupled with filtering, dilution, and ultimate release at stack height, would provide a significant "engineered safeguard."

The two proposed methods of disposal of the contaminated gas from the intermediate space can, of course, be combined; i.e., holdup as long as the activity is too high or the meteorological conditions for dispersal are unfavorable, and release whenever possible in order to extend the time after which release would be a necessity.

In any case the multiple-containment scheme described depends for proper functioning on the operation of the blowers that maintain the negative pressure in the intermediate space. These blowers have to function for a long time after the accident. Redundancy and the utmost reliability are required. Even if the blowers fail, however, some benefit will be derived from the multiple containment because the small leakage from the inner zone and the outer atmosphere into the intermediate zone will require some time to bring the intermediate zone to a positive pressure. It is clear that a large intermediate-zone volume and small rates of leakage into this volume are assets for a multiple-containment system.

Schemes other than multiple containment have been suggested for the reduction of radioiodine and other radioactive fission fragments. Among these are internal air circulation and purification and spray or chemical decontamination. What equivalent credit can be given to

these schemes has not yet been determined.

Specific Proposals for Reactors with Multiple Containment

The Consolidated Edison Thorium Reactor at Indian Point, Buchanan, N. Y., was the first large reactor designed to use the multiple-containment scheme.⁹ In this installation a "conventional" high-integrity containment vessel is surrounded by an additional concrete shell. The main purpose of the concrete shell was to provide shielding from the gamma rays emitted inside the high-integrity containment vessel by radioactive isotopes that would be released during the postulated maximum credible accident. There is an intermediate space between the high-integrity containment vessel and the concrete shell that can be operated at a negative pressure by exhausting air from it through a filter and a high stack. The filter is actually an "absolute" filter that was designed to retain particulate matter and not specifically iodine. When the distance graphs were developed long after the plant had been designed, however, it was found that the site met the requirements of low-population area and the population-center distance without credit for the multiple-containment feature. With respect to the exclusion area, the multiple-containment feature is called upon only to justify a safeguard factor of 2.

The N.S. *Savannah* also has a high-integrity containment vessel surrounded by the reactor compartment, which can be kept at negative pressure by exhausting air through a filtration system, including iodine filters.^{10,11} Inasmuch as a nuclear vessel is necessarily close to populated areas when in port, the multiple-containment features of the N.S. *Savannah* were intended to compensate for its inability to meet the standard site criteria.

Recently, the Department of Water and Power of the City of Los Angeles proposed to build a reactor of about 1500 Mw(t) on a site at Corral Beach about 32 km from Los Angeles.¹² (See a discussion of this proposal in Sec. VI, under "Action on Reactor Projects by Licensing and Regulating Bodies.") The population-center distance is 16 km, and the low-population zone radius is 8 km. The shortest distance to a residence is 170 m, but there are few residences closer than 700 m, and, if they do not

interfere with the exclusion area, the low-population zone would limit the reactor power to ~350 Mw(t) except to the extent that engineered safety features might reduce the distance requirements. It is proposed that additional reactors be constructed later at the same site.

Two reactor and containment concepts for the Los Angeles reactor have been considered by the staff of the AEC Division of Licensing and Regulation and by the Advisory Committee on Reactor Safeguards. One of these concepts provides concentric steel shells surrounding the primary system. The space between the shells is filled with "popcorn" concrete, a form of concrete having many interconnected voids but considerable strength. These voids act as the intermediate zone discussed above. Outside the outer of the steel shells is a thick layer of reinforced concrete, which gives the whole structure mechanical strength and also acts as shielding from direct radiation.

The second concept involves pressure suppression¹³ and a reasonably leaktight containment building. In the pressure-suppression system, a pool of water is provided in which the steam released by the postulated accident is quenched. The postaccident pressure is thus drastically reduced. The pool is always there, and there is no dependence on spray systems that would have to be started after the accident. Also, the intermediate zone, i.e., the space between the pressure-suppression system and the surrounding containment building, has a very large volume. The primary system extends outside this multiple-containment system, however, through a long steam pipe, the turbine, and the condenser. In the case of a nuclear excursion that would release high-pressure steam and radioactivity simultaneously, the isolation valves designed to shut off the containment vessel from the exterior primary piping would be required to close fast and tightly. Therefore some containment provision for the external part of the primary system has been proposed.

The Consolidated Edison Company of New York, Inc.,¹⁴ proposes to construct and operate a 2000-Mw(t) nuclear steam-generating unit at its Ravenswood site inside the New York metropolitan area. The population-center distance would be 100 m. In other words, almost complete reliance would be placed on engineered safeguards. The proposed reactor is of the

pressurized-water type and is to be housed in a structure similar to the concentric shells discussed above in connection with the Los Angeles reactor.

The Los Alamos Scientific Laboratory (LASL) and the Oak Ridge National Laboratory (ORNL) are considering the use of very large reactors, up to 160,000 Mw(t), as sources of process heat for the desalting of sea water.¹⁵ Assuming that the distance graphs could realistically be extrapolated to such high power levels, they would indicate, for a 100,000-Mw(t) reactor, an exclusion radius of 15 miles, a low-population zone radius of 215 miles, and a population-center distance of 288 miles. These distances are obviously prohibitive. Independent units of 25,000 Mw(t) could, however, be used, and, if double containment justified a safeguard factor of 20, the exclusion radius would shrink to 0.8 mile, the low-population-zone radius to 12 miles, and the population-center distance to 16 miles. An important feature of these reactors is that their output, the fresh water, is storable, and hence shutdown of the reactors could be tolerated more easily than by a producer of electric power; further, the reactors could run at constant power. This might result in simplifications of the control circuit; for instance, shutdown xenon would not have to be overridden and no rapid load changes would have to be followed.*

*The views expressed in this paper are not necessarily those of the AEC Advisory Committee on Reactor Safeguards of which Dr. Ergen is presently a member.

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Site Criteria for Reactors with Multiple Containment

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2.4 Effect of Engineered Safeguards on Reactor Siting

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An analysis of recent licensing trends provides some insight as to the acceptability and effectiveness of several engineered safeguards to reduce reactor siting distances. This article presents a tabulation of the water reactors in the United States, indicates the engineered safeguards provided for these reactors, and then associates an apparent engineered-safeguards factor for each reactor. Although the effectiveness of any engineered safeguard can be judged only on its application to a specific design, the author concludes that, on the basis of existing trends in the licensing of reactors, apparent engineered-safeguards factors of up to 20 can be achieved in an adequately designed reactor facility.

Engineered safeguards are safety features for preventing or limiting accidents or for mitigating the consequences of accidents. As applied to nuclear-power-reactor facilities, engineered safeguards include (in addition to containment structures) emergency cooling systems, filter systems used during an accident, containment-vessel sprays, and core sprays. The containment structure is the most widely accepted type of engineered safeguard, and the existing AEC Reactor Site Criteria¹ are based on the acceptability of such a device. As a guide to (1) how these criteria may be applied and (2) the factors of importance in the evaluation of a reactor site, the AEC issued a report² that describes the calculation of certain distance factors relative to the separation distance between the reactor and nearby populations. The distance factors define the "exclusion-area distance," the "low-population-zone distance," and the "population-center distance."

On the basis of the site criteria and the supplemental guide, the distance factors have been determined in Ref. 2, with the only influence being the power level. These distances are shown in Fig. 1, where the release of iodine and the resulting dose to the thyroid have established the controlling distances. The dashed part of the curve for the exclusion-area radius indicates that for low-power reactors the controlling distance is established by direct radiation rather than by leakage of iodine. The distance factors shown in Fig. 1 represent upper limits for a reactor site, inasmuch as the basis for a determination of the required distances involves assumptions that are extremely conservative. The AEC guide

Fig. 1 Reactor siting distance factors as a function of thermal power.

does not specifically rule out analyses based on less conservative assumptions; in fact, a report was issued recently by the Advisory Committee on Reactor Safeguards (ACRS) which discusses the applicability of engineered safeguards.³

The containment system is presumed to be effective when properly designed, leak tested, and maintained, but the acceptability of other engineered safeguards and the amount of credit that should be allowed for such safeguards have been points of debate for several years. In recent years a number of reactor installations have been licensed for operation on sites that do not meet the three distance criteria shown in Fig. 1. The reactor installations were licensed because they incorporated certain additional engineered safeguards for which the ACRS

^{*}Tennessee Valley Authority.

and the AEC Division of Reactor Licensing were willing to allow some credit. Thus, from an analysis of recent licensing trends, it is possible to obtain some insight as to acceptability and effectiveness of several engineered safeguards.

Applicable Engineered Safeguards

CONTAINMENT OR CONFINEMENT SYSTEMS

The distance factors shown in Fig. 1 are based on a containment system having the capability of limiting activity release to 0.1 Ci per day. Systems presently being used or proposed include

1. Single containment of the entire primary coolant system
2. Single containment of only a portion of the primary coolant system
3. Double containment
4. Confinement following pressure release

Reference 3 indicates that "when competently designed, constructed, maintained, and tested, containment and certain confinement systems are considered to be effective safeguards."

CONTAINMENT PRESSURE- AND TEMPERATURE-REDUCING SYSTEMS

Systems are included in reactor designs to reduce pressure and temperature following an accident where such reduction would decrease the activity leakage from the reactor building. Those presently being used or proposed include

1. Pressure-suppression systems
2. Spray systems within the containment vessel
3. Heat exchangers within the containment vessel
4. Energy release prior to containment of activity (used in conjunction with the confinement concept)

With the exception of item 4, these systems may be considered as acceptable safeguards if properly designed and if conservative assumptions are made regarding their effectiveness for reducing containment pressure and temperature.

AIR CLEANING SYSTEMS

Air cleaning systems directly reduce the amount of activity within the reactor building and also reduce the activity concentration in gaseous discharges from the reactor building. Credit is given for this safeguard in some cases when conservative lower bounds of removal efficiencies are used.

CORE SPRAY AND SAFETY-INJECTION SYSTEMS

Core spray and safety-injection systems are used to prevent core meltdown. Although these systems are usually considered essential to water reactors, credit is generally not given for these systems when they are the only engineered safeguards.

Evaluation Factor

The amount of credit given for engineered safeguards depends on a large number of interdependent factors which must be evaluated for a specific reactor design. Of particular importance, the reliability of engineered safeguards should be demonstrated by tests at suitable intervals under conditions similar to those during an accident. Table 1* lists the pressurized- and boiling-water power reactors in the United States and also gives the site distances associated with each reactor and those calculated from the site criteria. In general, it may be seen that the reactors presently operating meet the distance criteria. However, since many of these reactors were licensed prior to the issuance of the site criteria, the fact that they meet the criteria, with no credit being required for engineered safeguards, primarily reflects the fact that they are stations of limited size. On the other hand, it may be seen that the most recent power stations with thermal power levels ranging between 1200 and 1600 Mw could not meet the site distances shown in Fig. 1 without being subjected to rather serious economic disadvantages. The engineered safeguards for these reactors and the apparent engineered-safeguards factor, F , associated with each reactor are also listed in Table 1. These "apparent" engineered-safeguards factors, F , are obtained when the power level that has been authorized (or applied for) for a given site is compared to that obtained from Fig. 1 for the actual site distances.

Table 1 shows that apparent engineered-safeguards factors ranging from 1.5 to 23.4 exist for reactors presently authorized for construction or for which construction-permit applications have been made. These factors do not necessarily represent the true value of the safeguards provided for a particular plant but rather are a measure of what was needed for the particular reactor at the given site. In a like manner, the fact that a construction permit has been granted for a plant with an apparent engineered-safeguards factor of approximately 20 does not necessarily imply that such a factor is warranted for any reactor at any site. An examination of the apparent safeguards factors listed in Table 1 for specific reactors, the engineered safeguards associated with these reactors, and statements made by the ACRS do, however, provide some insight into what kinds of factors appear to be reasonable and the requirements of a design to obtain such factors. Nine reactors listed in Table 1 would not meet the basic site criteria unless credit were given for engineered safeguards. Four of these reactors—Indian Point, Shippingport, Humboldt Bay, and Pathfinder—were authorized for construction before the site criteria were issued, and therefore their ap-

*For reference convenience the reactors are listed in alphabetical order in Table 1.

Table 1. DISTANCE FACTORS FOR SITING U. S. POWER REACTORS
(Reactors Listed in Alphabetical Order)

Reactor name	Reactor type	Thermal power, Mw	Exclusion-area radius, miles		Low-population-zone radius, miles		Population-center radius, miles		Approximate population†	Engineered safeguards	Engineered-safeguards factor, $\frac{1}{F}$	Reactor status
			Calculated*	Actual	Calculated*	Actual	Calculated*	Actual				
Big Rock Point	Boiling water	240	0.278	0.31	3.91		3.21	138.0	18,300†	Single steel containment vessel; spray system within vessel; core spray system		Received 40-year license in May 1964
Connecticut Yankee	Pressurized water	1473	0.87	0.28	13.38		17.81	8.5	33,250	Single reinforced-concrete container with steel liner; internal recirculation spray system within containment building; continuously operated air-recirculation system with cooling and filter-adsorption units; borated-water safety-injection system	8.84**	Granted construction permit in May 1964
Dresden	Boiling water	700	0.34	0.50	9.08		10.77	14.0	51,401	Single steel containment vessel; building water-spray system		Initially critical in October 1959
Humboldt Bay	Boiling water	240	0.278	0.13	3.88		5.20	4.0	28,137	Pressure-suppression system; core spray system	3.0**	Power level increased from 165 to 240 Mw(t) in May 1965
Indian Point	Pressurized water	816	0.81	0.30	7.70		9.70	17.0		Double containment	2.15	Reached full-power operation in January 1963; power level increased from 585 to 816 Mw(t)
LaCrosse	Boiling water	185	0.219	0.21	3.11	3.0	4.15	19.0	56,000	Single steel containment vessel; building spray system (manual); core spray system		Construction authorization issued in March 1963
Malibu	Pressurized water	1473	0.87	0.11	13.38	4.97	17.81	9.94		Double containment; spray system within containment building; borated-water safety-injection system; air recirculation through coolers	23.4**	Construction permit applied for in November 1963
Nine Mile Point	Boiling water	1538	0.88	0.76	13.67		18.23	7.0	22,133	Pressure-suppression system; core spray system	4.1	Construction permit granted in April 1965
Oyster Creek	Boiling water	1600	0.90	0.25	13.88		18.64	10.0	25,000	Pressure-suppression system; core spray system	7.44**	Construction permit granted in December 1964
Pathfinder	Boiling water, nuclear superheat	190	0.237†	0.50	3.35		4.45	3.5	85,466	Single steel containment vessel; core spray system	1.43	Authorized to operate up to 1 Mw(t) in March 1964
PWR, Shippingport	Pressurized water	505	0.44	0.40	8.5		8.85	7.5	25,000	Single steel containment vessel; core spray system	1.22	Operating since 1957; power level increased from 231 to 505 Mw(t)
San Onofre	Pressurized water	1219	0.78		11.91		15.75	2.0	24,000	Single steel containment vessel; spray system within reactor building; air cleanup system; within containment vessel; borated-water safety-injection system	20.9	Construction permit issued in March 1964
Yankee	Pressurized water	800	0.49	0.50	7.47		9.89	21.0	21,587‡	Single steel containment vessel; air-recirculation system; borated-water safety-injection system		Operating since 1960; granted license amendment in December 1963 for operation up to 800 Mw(t)

*Calculated in accordance with criteria of Ref. 1.

†Approximate population at actual population-center distance or at distance given.

‡Ratio of the authorized (or proposed) power level to the power-level limit obtained from Fig. 1 for the actual site distances.

§Calculated as 0.30 mile based on gamma dose.

¶At 45 miles.

**Gamma dose neglected.

††Calculated as 0.29 mile based on gamma dose.

‡‡At 9 miles.

parent safeguards factors are not pertinent. It may be noted, however, that increases in power level have been authorized for three of these reactors. Two of the plants—Nine Mile Point and Oyster Creek—are boiling-water reactors that use the pressure-suppression system for reducing pressure and activity release. The apparent safeguards factors for these plants are 4.1 and 7.4. The other three reactors are pressurized-water reactors that have either single or double containment in conjunction with other safeguards to reduce activity release. In the case of the Connecticut Yankee plant, a safeguards factor of about 7 is indicated, whereas in the case of San Onofre a factor of approximately 21 is cited. Each of these plants has single containment and other engineered safeguards. In the case of the San Onofre reactor, it is particularly interesting to note that the ACRS report⁴ indicates that approval was based not only on a number of engineered safeguards, including a safety-injection system, but, in addition, the favorable meteorology at the site. Although safety-injection systems are generally required for water-cooled reactors, credit had not previously been granted for this engineered safeguard.

One of the proposed pressurized-water reactors, Malibu, has double containment. The engineered-safeguards factor for Malibu reactor is 23.4. The case of the Malibu plant is of interest since it provides some basis for an evaluation of the factors that can be used for a pressurized-water reactor with double containment. Two letters issued by the ACRS in the fall of 1962 relating to the Malibu reactor site are of significance. Portions of these letters, as discussed in Ref. 5, indicate that the apparent engineered-safeguards factor for the double-containment concept is significantly higher than for the pressure-suppression system. Since the application for a construction permit for the Malibu plant has not been approved, it cannot be stated conclusively that a factor of 23.4 has been allowed. However, the preliminary report issued by the ACRS indicates their approval of the proposed site with the appropriate safeguards.

As the power levels of reactors increase beyond those indicated in Table 1, engineered-safeguards factors in excess of those listed will be required to limit the site distances to reasonable values. The distances given are dictated primarily by the release of iodine. Therefore engineered safeguards directed at reducing iodine release are most effective in reducing the distance factors. However, in an evaluation of specific engineered safeguards and their effect on reducing site distances, careful consideration must also be given to the release of particulate and noble-gas activity and the whole-body

dose resulting from such release. Ergen⁶ has shown that iodine release and the dose to the thyroid are the controlling factors in establishment of the exclusion-radius distance until the iodine release is reduced by a factor of 22.9. Further reduction in iodine release would not result in an improvement of site distances unless there were adequate reductions in the release of noble gases. Ergen⁶ also shows that, in establishment of the low-population-zone distance, the iodine release and thyroid dose are controlling until the iodine release is reduced by a factor of 153, at which time the whole-body dose is controlling.

Summary and Conclusions

The AEC site criteria may be used as a guide in selection of a site prior to the development of the detailed design of a nuclear power station, and specific engineered-safeguards factors may then be considered as means for reducing certain distance factors. The large number of features that are pertinent in evaluation of the adequacy of a specific reactor design at a particular location can be considered only when the detailed design has been completed.

Engineered-safeguards factors ranging between values of 7 and 25 appear to be reasonable for planning purposes at the present time. In the use of these numbers, it should be recognized that they are achievable only when certain engineered-safeguards systems are included in a reactor design. The determination of allowable safeguards factors based on detailed design features should be an early objective in any reactor project.

Present research and development, as well as accumulated operating experience, if favorable, may be expected to result in greater credit for engineered safeguards, and thus a much needed relaxation will be permitted in the required siting distances for future large-sized nuclear power units.

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2.5 Siting of Power Reactors

Harold L. Price

[*Nucl. Safety*, 9(1): 1-4 (January-February 1968)]

[Editor's Note: Harold L. Price is Director of Regulation for the USAEC. One of his primary responsibilities in that position is the safe siting of nuclear power reactors. *Nuclear Safety* has had a continuing interest in the problems and challenges of siting and has carried some two dozen articles on the topic, starting with two in its very first issue (cf. *Nuclear Safety*, Vol. 1, No. 1, pages 4-5 and 40-42, September 1959). In the preceding issue we carried a lead article by F. R. Farmer, UKAEA, on proposed siting-risk criteria. The following article presents the AEC position on siting problems, as outlined by Mr. Price at the 12th Annual Health Physics Society Meeting at Washington, D. C., on June 20, 1967.]

Abstract: This article discusses AEC regulations regarding reactor siting and sets forth the AEC position on the subject at this time. Factors affecting reactor siting are detailed in Part 100 of the Code of Federal Regulations, Title 10. They include characteristics of reactor design, population density, and physical conditions of the site. Even where unfavorable site characteristics exist, however, the site may be judged acceptable if adequate compensating engineered safeguards are included in the design. There are two major areas related to power-reactor siting that have proved to be the most troublesome: metropolitan siting and the location of reactors in areas of high seismicity. The following are providing help in resolving the safety questions involved: Codes and specifications are being improved, more reliable engineered safety features are being designed, research and development work is under way, experience in the operation of power reactors is being gained, and the development of more definitive criteria is planned or being considered. Meanwhile, it is not likely that a large power reactor will be located in the midst of a densely populated metropolitan area or on a major active fault such as the San Andreas.

There are many factors that affect the siting of power reactors in the United States. The U. S. Atomic Energy Commission (AEC) has no responsibility and no authority to deal with a number of them since the Commission's regulatory authority is limited essentially to matters of radiological health and safety and the common defense and security. These factors over which the AEC has no authority include aesthetics, zoning, cost and availability of land, and thermal effects of cooling-water discharges. However, there are many safety-related factors affecting the siting of power reactors that are within the responsibility and the authority of the AEC. These factors are described in Part 100 of the AEC regulations, the site-criteria regulation.

Part 100 broadly groups the site evaluation factors considered by the AEC into the following categories:

1. *Characteristics of Reactor Design.* This category includes the proposed use and maximum power level of a reactor, the extent to which generally ac-

cepted engineering standards are to be used in its design and construction, the extent to which it will incorporate unique or unusual features having a bearing on safety, and the engineered safety features proposed for the reactor to reduce the likelihood and consequences of reactor accidents.

2. *Population Density.* This category includes the amount of land under the control of the reactor owner, the distance to the nearest boundary of a densely populated center, and the total number of people and density of population between the reactor site and the population center.

3. *Physical Characteristics of the Site.* These characteristics include seismology, meteorology, geology, and hydrology.

The Principle of Compensating Safeguards

Part 100 also states an important principle that has guided the Commission in the application of this regulation since its publication in 1962. This principle is that, where unfavorable characteristics of the site exist, the proposed site may, nevertheless, be found acceptable if the design of the facility includes appropriate and adequate compensating engineered safety features. These site characteristics may include both those related to population density and distribution in the site vicinity and those related to physical characteristics of the site, such as meteorology or seismology.

Since the site criteria were published in 1962, this principle has been recognized and applied in the review and approval of construction permit applications. Engineered safety features have been accepted as partial compensation for distance to people. Consequence-limiting safety features, such as improved emergency core-cooling systems, iodine filtration systems, containment penetration pressurization systems, containment cooling systems, and double containment, have importantly influenced the approval of many of these projects. The generally favorable experience since 1962 with the operation of water power reactors has been a related and by no means less important factor.

The Commission also has applied the principle of compensating for unfavorable site characteristics with suitable engineered safety features in cases where the unfavorable characteristics related to physical conditions of the sites, such as meteorology, hydrology, and seismology. The safety features that were pro-

vided to compensate for these types of site characteristics generally consisted of accident-prevention features.

For example, in reactors located along the East Coast, such as the Florida power reactors, the designs include provisions to prevent malfunctions that might result from the high waters and winds from hurricanes. In Midwest locations, power reactors such as the Dresden reactors being constructed by Commonwealth Edison include multiple transmission lines from off-site power sources, an on-site emergency power source, and concrete protection of exposed portions of the reactor important to public safety in order to protect against tornadoes. The designs of power reactors located in seismic areas—particularly in West Coast locations—include provisions to withstand the ground vibrations associated with earthquakes that might be anticipated to affect the sites at which they are located.

Recent Developments in Site Evaluation Factors

I will now return to the broad classifications of site evaluation factors given in Part 100 to discuss the more recent developments in each area.

Industry practice, as approved by the Commission, has been to locate large power reactors in areas of relatively low population density so that evacuation or other protective measures can be taken in an emergency and at some distance from large population centers where such protective measures would be difficult. In such locations and under weather conditions prevailing most of the time, the need for stringency in control of routine effluents has been reduced, and the dilution and dispersion characteristics of the atmosphere have provided a high degree of protection against consequences from serious accidents.

We have also required that power reactors be provided with engineered safety features designed both to prevent accidents and to limit the consequences of accidents. The distance requirements have provided further assurance of safety by providing a valuable element of time in an emergency during which measures to protect the surrounding population could be taken.

At present there are trends in the nuclear power industry that have a significant bearing on the future applicability of this practice. Power levels of proposed reactors are becoming higher and fuel cycles longer, and some of these reactors are being proposed for construction closer to large metropolitan centers. The increase in reactor power and the lengthening of the fuel cycle in terms of percentage fuel burnup increase the fission-product inventory accumulated in the fuel of the reactor.

On the positive side, advances in technology con-

tribute increased safety in the design and construction of reactors.

In light of these developments, an important reactor siting question for the future is: How much emphasis should be placed, relatively speaking, on (1) distance from population centers, (2) engineered safety features designed to prevent accidents, and (3) engineered safety features designed to limit their consequences? This question applies to all sites.

The Question of Metropolitan Sites

The basic safety issue presented by metropolitan sites for large power reactors is whether, in lieu of partial dependence on distance from population centers, full reliance can or should be placed on the inherent safety of the reactors and their engineered safety features.

Testimony at the recent regulatory hearings before the Joint Committee on Atomic Energy gave the best answer we now have to this question. Chairman Glenn T. Seaborg presented the Commission's conclusions and policy on this matter as follows: "While progress has been made in improving the safety of reactors, we need to obtain more experience and data in order to develop criteria for the siting of these units in urban areas."

Commissioner James T. Ramey, testifying before the Joint Committee in 1965, and again in 1967, concerning the question of metropolitan siting stated that further important advances were needed in reactor plant design, particularly in the capability of safety systems and engineered safety features and in adapting these critical systems so that they could be inspected and tested. He also expressed the need for practical demonstrations of the dependability of these systems.

This accumulation of experience could come not only from the operation of the larger reactors now under construction but also from mockup tests and experiments, engineering analyses, and in-place tests in these operating reactors.

Until these data and further research and development results are available, we will continue to rely on both low-population-density requirements and engineered safety features to assure adequate protection to public health and safety, and our siting policy for large power reactors will continue to reflect both of these considerations. We will seek to maintain a proper balance between both kinds of engineered safety features—those which are designed to limit the consequences of reactor accidents and those which are designed to prevent their occurrence, such as improved quality assurance.

Safety Research Programs

The safety questions that need to be resolved are

being attacked from many different directions. Much effort is currently being placed on improving and upgrading codes and standards for reactor components and on the development of criteria for reactor design. Large safety research programs are in progress, and engineering improvements are being worked out by manufacturers in the design of their reactors.

To encourage and foster improvements in the quality of reactor components and systems, the AEC's developmental staff has embarked on a large-scale program to enhance the quality of components to be used in Commission-owned reactors. Detailed specifications and requirements are being tabulated and later will be made available to all reactor designers so that they may take advantage of the progress made in better defining the quality of components to be used in AEC reactors.

In addition to these steps, the AEC's regulatory staff has held discussions with industry code committees—for example, the ASME Code Committee—concerning the generally recognized need for greater assurance of integrity in the primary systems of reactors. The ASME Code Committee has undertaken an extensive reexamination of the code for nuclear pressure vessels. In addition, the activities of the Code Committee for nuclear pressure piping have been stepped up, and a committee has been assigned to develop codes for the pressure integrity of primary-system pumps and valves.

Further, in January 1967 a joint effort was undertaken by the Advisory Committee on Reactor Safeguards and the Commission staff to develop detailed regulatory requirements for the design, fabrication, inspection, and testing of nuclear pressure vessels and for other portions of the reactor coolant system that go beyond present code coverage toward a higher level of integrity and reliability. We expect that initial portions of this task will be completed this year. The work of this group is in parallel with that of the ASME Code Committee to which I referred earlier, and coordination between the two groups is being maintained.

To increase the capability and effectiveness of consequence-limiting safety features, the Commission has undertaken an extensive research and development program to define better, through experiments and analyses, those accident situations with which the engineered safety features of a large power reactor must cope. This research and development program includes the Loss-of-Fluid Test (LOFT) project, which will experimentally determine the effects on a reactor core of a sudden loss of coolant, and numerous other projects at the national laboratories that are related to fission-product release and transport, metal-water reactions, and containment leakage and integrity experiments. In addition, specific projects are directed toward determining through experiments and tests the effectiveness and reliability of current emergency core-cooling systems and other important safety features of reactors.

Similarly, reactor designers are responding to the challenge to develop simple and more reliable engineered safety features for power reactors. The nuclear industry has a major role to play in carrying out the safety research necessary to keep pace with the development of the nuclear power industry, and the Commission has invited the industry to expand its efforts in this field.

Until the further developments and experience, which I have briefly described, become available, I would expect that the past practice of locating large power reactors outside of densely populated metropolitan centers will continue. This is in keeping with our present site criteria. Although the regulations do not offer definitive guidance on what constitutes a densely populated metropolitan center, the industry has considerable guidance based on the past application of the general site criteria to individual sites that have been approved.

Physical Site Characteristics

The third and last broad category of power-reactor siting factors given in Part 100 includes those relating to the physical characteristics of a site. Current and prospective developments in this area are worthy of mention.

Our experience has shown that of all the "ologies" which define the physical characteristics of a site and have a bearing on reactor safety, geology and seismology have been the most troublesome. This is particularly true of West Coast locations where the probability and amount of permanent ground displacement due to earthquakes must be considered. Efforts of the Commission to deal with this reactor siting consideration have been proceeding along two lines.

One of these efforts involves research work to develop and test design concepts and methods for power reactors to protect against the effects of permanent ground displacement beneath a reactor facility. The Oak Ridge National Laboratory is directing this project as a part of the Commission's safety research program. Its goal is to develop conceptual designs of reactor plants which would accommodate moderately severe seismic effects, including permanent ground displacement. The value of model testing of some of the more promising concepts is also being considered.

We believe also that definitive guidance beyond that presently given in Part 100 concerning the seismological and geological factors that affect power-reactor site selection would be desirable. Over the past several months we have been developing additional guidance for prospective applicants in this area. Our initial efforts have been toward developing criteria for determining the maximum ground acceleration at a given location for design purposes, for identifying an active fault—that is, a fault that might be expected to be the locus of permanent ground dis-

placement in the event of a seismic disturbance, for establishing a zone adjacent to such a fault in which such ground displacement should be considered in reactor design, and for defining the amount of permanent ground displacement which would be a design criterion for a power-reactor facility located within the zone. Our earth science consultants from the U. S. Coast and Geodetic Survey and U. S. Geological Survey have been assisting us in this work.

We plan to develop criteria for the design of reactor facilities that are proposed to be located in seismic areas. These engineering criteria will describe the procedures that may be used in seismic design analyses in a general way without prescribing details of procedure.

Summary

In summary, there are two major areas related

to power-reactor siting which have proved to be the most troublesome. These involve metropolitan siting and the location of reactors in areas of high seismicity. On the basis of present information, it is not likely that in the near future a power reactor will be located in the midst of a densely populated metropolitan area, and a plant of this type will certainly not be located on a major active fault such as the San Andreas. The questions that remain are: How close to such locations can power reactors be built and under what conditions? As an aid in resolving the safety questions involved, codes and specifications are being improved, more reliable engineered safety features are being designed, research and development work is under way, experience in the operation of power reactors is being gained, and the development of more definitive criteria is planned or being considered.

2.6 Siting Practice and Its Relation to Population

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[*Nucl. Safety*, 14(6): 576-584 (November-December 1973)]

Abstract: A study was made to determine the relation between power-reactor sites and the surrounding population in the United States and in some foreign countries. Results show that the population-vs.-distance envelope represented by a combination of the Indian Point and Zion sites has served as an upper boundary in current siting practice. It is also shown that the application of engineered safety features has had no discernible effect on the selection of sites.

In the Feb. 1, 1973, issue of the *Federal Register*,¹ the U. S. Atomic Energy Commission (AEC) announced its intent to initiate a program for the development of general environmental siting criteria for nuclear power plants as another step in the evolution of criteria to provide guidance for the site-selection process. This program will advance previous efforts to establish siting criteria for power plants, especially nuclear power plants, the results of which appear in the *Code of Federal Regulations*,² in special reports of the federal government,³⁻⁶ and in the proceedings of a recent international symposium.⁷

The growing demand for new power stations, many of which will be nuclear, requires the selection of many new sites; this in turn requires long-range planning, not only for the use of sites to support single power plants but also for larger sites that may support several plants. Surrounding population is one of the most important criteria in the siting of nuclear power plants, particularly when the low-population zone is small or the population density is high, or both. This article assesses current nuclear power-plant-siting practice as it relates to the surrounding population. The subject is discussed in terms of past practice and stated AEC policy, the object being to present some insight into the practice and trends of siting that would be of interest to those who must confront the problem.

REGULATIONS

The siting regulations established so far are published in Title 10 of the *Code of Federal Regulations*,²

*Oak Ridge National Laboratory.

Part 50 contains the basic information relative to licensing of production and utilization facilities. Part 100 is particularly applicable to the various problems involved in site selection. The purpose of Part 100 is to set forth the present criteria used in the evaluation of the suitability of a proposed site. In addition to those basic criteria, Part 100 gives reference values for dose limits and defines the exclusion area, low-population zones, and population-center distance.

USAEC Report TID-14844 is of historical significance to siting but is no longer actively used⁹; Regulatory Guides 1.5 and 1.4 are presently the primary references used in siting studies.^{9,10} These guides are intended to provide help in calculating the various distance requirements for reactor sites by developing examples in which conservative assumptions concerning fission-product escape from fuel, removal

Initial U. S. Practice

The earliest reactors were constructed for research, test, and demonstration of the feasibility of nuclear-energy uses and, as such, were under direct AEC control and were not subject to licensing under 10 CFR 50 procedures. As nuclear power emerged as an infant industry, the first generation of utility-type power reactors, which were licensed according to procedures outlined in 10 CFR 50, included Dresden (45 miles from Chicago, Ill.), Yankee (40 miles from Springfield, Mass.), and Indian Point (35 miles from New York City). Construction permits for, and operation of, these plants followed the schedule in Table 1.

For consistent comparisons between sites, graphs have been made that show the cumulative populations for given distances from the reactor plants. The

Table 1 Construction and Startup Information for the First Industrial U. S. Power Reactors

Plant	Original design power, MW(t)	Construction permit		Completion date	Date critical	Date at power
		Number	Date			
Indian Point	585	CPPR-1	5-4-56	4-62	8-62	12-62
Dresden	630	CPPR-2	5-4-56	9-59	12-59	6-60
Yankee	400	CPPR-5	11-4-57	7-60	8-60	12-60

from the containment atmosphere, release to the environs, and downwind dispersion are made for the accident condition.

In the application of these regulations, subjective judgments must be made by the regulatory authorities. These judgments and various statements made by those in policy-making positions become part of the basis for making a siting decision. In general, revisions in the practice of siting have been made patiently and deliberately and have been based on siting experience at existing plants.

PRACTICE

During the past 18 years, many reactors have been designed and constructed for the production of commercial electric power, and operating experience has been gained in the United States and in several foreign countries. The location of reactors is of particular interest because of the possible effects of the release of their fission products on the health and safety of the general public.

population distributions for these initial three plants are shown in Fig. 1. The Dresden and Yankee sites are nearly identical with respect to population, but Indian Point is closer to a greater number of people than either of the other two plants. From the preliminary safety-analysis reports (PSARs), it appears that the construction permits for all three plants were granted on approximately the same basis, that is, the reactors were to be surrounded by high-pressure containment vessels; but, of course, they did not include many of the more sophisticated engineered safety features developed since then that are required in the large high-power reactors of today. The containment vessel with its appurtenances and the reactor shutdown system are illustrations of devices for minimizing the probability and/or effect of a power-plant accident. It was in the early sixties that these devices became known as engineered safety features, and prescribed allowances were made for the performance of these devices in calculating the resulting dosage due to assumed reactor accident conditions.

Construction of the Dresden and Yankee plants

Fig. 1 Cumulative population vs. distance for the first utility-type power reactors; —, data from the safety-analysis reports normalized to 1960; - - - -, estimated from 1960 data.

followed the best power-plant practice at that time. In addition, the Indian Point plant incorporated some unique and expensive engineered safety features. Whether or not this emphasis on extensive safety features was required by the AEC Division of Licensing and Regulation is not specified in the licensing documents. However, Consolidated Edison apparently realized that the population density was higher near the Indian Point site and used engineered safety features to assure that fission-product releases to the environs in the unlikely event of an accident would be small. In so doing, Consolidated Edison employed the concept of careful attention to design and construction and the use of unique safety features that would allow close-in siting.

Current U. S. Practice

Cumulative populations at various distances from the plants are given in Table 2 for a number of U. S. light-water-cooled reactor plants for which formal

applications for construction permits have been made and for some foreign plants for which information was available. The population distributions of selected plants are plotted in Fig. 2.

Of the plants designated in Fig. 2, the first to receive a construction permit (May 4, 1956) was Indian Point. In the subsequent 13 years, a large number of plants were proposed, and, for many of these, applications for construction permits were filed. More than 50 construction permits were granted, but the population distributions at all approved sites were below that for Indian Point on the population-vs.-distance curve, even though some sites for which applications were filed had population distributions exceeding that at Indian Point. This situation was finally modified with the granting of a construction permit for the Zion station on Dec. 2, 1968, and will be further modified if the Newbold Island site is approved. Figure 2 shows that the population associated with Zion is slightly greater than that of Indian Point to about 13 miles out, and, from there on, it is well below that of Indian Point.

Sites That Were Dropped. Ravenswood (Consolidated Edison), Edgar (Boston Edison), Bolsa Island (Southern California Edison and San Diego Gas & Electric), Burlington (Public Service Electric & Gas), Bodega Bay (Pacific Gas & Electric), Easton (Niagara Mohawk), and Bell (New York State Electric & Gas) were either proposed and dropped before formal consideration or formally considered and then dropped. Bodega Bay was dropped because of its nearness to the San Andreas earthquake-fault zone, Easton because of anticipated adverse thermal effects and possible infringement on a national landmark, and Bell because of public sentiment against possible thermal alteration of Lake Cayuga.

Ravenswood was proposed on Dec. 10, 1962, for location in the Queens Borough of New York City,¹¹ but, after about a year of public activity, the proposed was dropped. Figure 2 shows the population distribution of Ravenswood compared with the distributions at other sites. Boston Edison considered some sites near Boston (e.g., Edgar) before going out of its own service area to purchase the Pilgrim site.¹² Since no particular site was ever selected, no realistic comparative curve can be drawn, but it is evident that the population distribution would have been beyond the Indian Point curve at least out to 10 or 15 miles. The Burlington site, which was abandoned, is an interesting example of siting experience. Extensive siting studies were made, and a great effort was expended to convince the people

of the small town near the site selected that the plant would present no undue hazard. However, after application was filed, the AEC advised the utility to select a site that was less sensitive as far as population distribution was concerned.¹³ Figure 2 shows clearly that Burlington's curve is beyond that of Indian Point out to about 35 miles. The Bolsa Island site presents a different story. The population-distribution curve goes beyond the Indian Point-Zion-Newbold envelope, but it was to have been an offshore site with a $\frac{3}{4}$ -mile exclusion distance provided by the ocean. The effect of the population distribution on approval of the site had not, however, been considered at the time. The project died because of increasing costs and lack of financial support.

Recent Actions. The Midland plant, proposed in October 1963 (formal application was made in January 1969, and the construction permit was issued Dec. 15,

1972), has drawn some attention with regard to siting and population. The AEC clearly stated that, since the Midland site is similar in population distribution to the Zion and Indian Point sites, the plant should have safety features at least equivalent to those of the Zion and Indian Point plants.¹⁴ The permanent Midland* population distribution exceeds the distribution at Zion and Indian Point at less than 3 miles, as shown in Fig. 2. It is significant to note that the safety features requested by the AEC are not the major engineered safety features, such as containment spray systems, emergency cooling systems, and safety-injection systems, which the plant already has, but are such features as means to accommodate reactor vessel movement, failed-fuel-detection equipment, accommodation of thermal shock after a loss-of-coolant accident,

*A Dow Chemical Company plant, with its large transient population (work force), is situated between the nuclear plant and the town of Midland.

Fig. 2 Cumulative population vs. distance for selected U. S. plants. Numbers in parentheses represent the year on which information was based. —, plants built or for which construction permits have been granted. . . . ., plants for which construction-permit applications are still pending. - - - -, plants for which construction-permit applications were officially made but subsequently withdrawn.

Table 2 Population Vs. Distance for Light-Water-Cooled Power Reactors

Reactor plant	Nearest city	Docket No. and date	Construction permit/operating license	Thermal power, MW	Reactor type and manufacturer	Population, (thousands)										Year of census
						0-1 miles	0-5 miles	0-10 miles	0-20 miles	0-30 miles	0-40 miles	0-50 miles	0-60 miles			
Arnold	Cedar Rapids, Iowa	331, 11-68		1670	BWR (GE)	0.015	2.73	75	172	228	343	537		70		
Beaver Valley	East Liverpool, Pa.	334, 1-69		2600	PWR (W)	0.72	1.74	174	519	1,860	3,195	4,322		70		
Big Rock Point	Ithaca, N. Y.	316, 3-68	Tabled	2436	BWR (GE)	0.116	9.84	31	144	419	1,081	2,023		60		
Boiling Bay	Troy, N. Y.	315, 3-68	5-60/8-63	1400	BWR (GE)	0.005	0.9	2.1	48	115(25)	21	131		60		
Boston Bay	Santa Rosa, Calif.	303	Withdrawn	1408	BWR (GE)	0	0.3	2.1	48	115(25)	21	131		60		
Brazos River	Long Beach, Calif.	307	Tabled	3400	BWR (GE)	0.59	1.01	812	3,101	5,883	7,880	8,916		70		
Brown's Ferry 1	Dexter, Ala.	324	Tabled	3400	BWR (GE)	0.33	2.76	22	100	309	412	523		60		
Brown's Ferry 2		340	5-67	3193	BWR (GE)											
Brown's Ferry 3		340	5-67	3193	BWR (GE)											
Brunswick 1	Wilmington, N. C.	324, 7-68	7-68	3193	BWR (GE)	0.144	3.51	7.39	73	117	140	182		60		
Brunswick 2		325, 7-68		3436	BWR (GE)											
Burlington	Philadelphia, Pa.	317, 1-68	Withdrawn	3250	PWR (W)	4.73	119	236	3,093	5,067	6,042	7,000		60		
Cabot CRRs 1	Washington, D. C.	318, 1-68		3450	PWR (GE)	0.17	11	52	180	372	1,033	4,710		61		
Cabot CRRs 2		318, 1-68		3450	PWR (W)	0.046	7.08	46	157	503	672	1,067		61		
Cook 1	Boston Harbor, Mich.	316, 12-67		3150	PWR (W)											
Cooper	Liaochu, Nehr.	298, 7-67	6-68	3381	BWR (GE)	0.004	4.18	3.49	37	97	104	178		60		
Connecticut Yankee	Middletown, Conn.	213	5-64/6-67	1973	PWR (W)	0	7.92	53	368	752	87	141		60		
Crystal River 3	Gainesville, Fla.	302, 8-67	9-68	2432	PWR (BW)	0	0.08	5.18	11	22	87	141		61		
Crystal River 4		303		2432	PWR (BW)	0	0.08	5.18	11	22	87	141		61		
Davis-Besse	Toledo, Ohio	346, 8-69		3433	PWR (BW)	0.008	2.22	15.40	95	617	882	1,815		60		
Diablo Canyon 1	San Luis Obispo, Calif.	175	4-68	3250	PWR (W)	0	0.81	1.37	49	87	122	149		60		
Diablo Canyon 2		323, 6-68		3250	PWR (W)	0	0.81	1.37	49	87	122	149		60		
Diablos 1	Joliet, Ill.	327	5-56/9-63	2700	BWR (GE)	0.1	2.3	38	187			5,857		60		
Diablos 2		327	5-56/9-63	2700	BWR (GE)	0.1	2.3	38	187			5,857		60		
Diablos 3		327	5-56/9-63	2700	BWR (GE)	0.1	2.3	38	187			5,857		60		
Edison	Troy, N. Y.	341, 4-69	Withdrawn	2181	BWR (GE)	0.234	4.46	42	497	655	844			60		
Fernald 2	Detroit, Mich.	341, 4-69		3253	BWR (GE)	0.137	6.83	50.91	326	2,137	3,485	4,300		60		
Fort Calhoun	Omaha, N. Y.	333, 12-68		2434	BWR (GE)	0	1.98	32	76	140	527			60		
Fort Calhoun	Omaha, Nehr.	385		1430	PWR (CE)	0.45	13	34	261	708	800			60		
Gina (Brookwood)	Rockledge, N. Y.	344	6-68	1300	PWR (W)	0.35	7.70	34	629	932				70		
Hatch	Waycross, Ga.	321, 5-68		2436	BWR (GE)	0.049	0.86	5.99	43	92	140			72		
Humboldt Bay	Eureka, Calif.	133	11-60/8-62	2400	BWR (GE)	1.7(1.5)	38	49	81	60(35)				70		
Hutchinson Island	Fl. Pierce, Fla.	335, 1-69		2700	PWR (CE)	0.1(2)	1.92	43	87(33)	1,720(23)	4,344(55)	12,324(45)	16,088(55)	60		
Indian Point 1	Poughkeepsie, N. Y.	347	5-56/2-62	3755	PWR (W)	1.02	53	155	277(15)					60		
Indian Point 2		347	10-66	3755	PWR (W)									60		
Indian Point 3		347	10-66	3755	PWR (W)									60		
Indian Point 4		347	10-66	3755	PWR (W)									60		
Indian Point 5		347	10-66	3755	PWR (W)									60		
Kewaunee	Mankato, Wis.	342, 6-69		3293	BWR (GE)	0.02	1.00	12	89	222	374			61		
Limerick 1	Philadelphia, Pa.	352, 3-70	6-68	1650	PWR (W)	0.02	65.93	129.4	691.6	3,497	5,282	6,324		60		
Limerick 2		352, 3-70		3293	BWR (GE)	0.02	65.93	129.4	691.6	3,497	5,282	6,324		60		
Limerick 3		352, 3-70		3293	BWR (GE)	0.02	65.93	129.4	691.6	3,497	5,282	6,324		60		
Lucas Yankee	Lawton, Maine	309	10-68	2440	PWR (CE)	0.54	2.85	26	49	220	417			60		
Malibu	Los Angeles, Calif.	314	Tabled	2432	PWR (W)	0	6	12	700	2,100				60		
Midland 1	Midland, Mich.	329, 1-69		2432	PWR (BW)	0.08	41	51	264	419	520	931		60		
Midland 2		330, 1-69		2432	PWR (BW)	0.08	41	51	264	419	520	931		60		
Midland 3		330, 1-69		2432	PWR (BW)	0.08	41	51	264	419	520	931		60		
Millstone Point 1	New London, Conn.	342	5-66	2810	BWR (GE)	0.198	67(6)	66	236					61		
Millstone Point 2		336, 2-69		2810	BWR (GE)	0.198	67(6)	66	236					61		

Monticello	St. Cloud, Minn.	263	6-67	1469	BWR (GE)	0.023	3.94	9.71	43	190	946			40
Newbold Island 1	Philadelphia, Pa.	354, 2-70		3400	BWR (GE)	4.9(2)	92	446	1,153	3,725	5,347	6,949	11,979	60
Newbold Island 2		353, 1-70		3400	BWR (GE)									
Nine Mile Point	Quebec, N. Y.	220	4-65	1538	BWR (GE)	0	1.98	32	76	140	527			60
North Anna 1	Fredericksburg, Va.	338, 3-69		2653	PWR (W)	0.012	1.04	9.24	44	184.3	573	879		60
North Anna 2		339, 3-69		2653	PWR (W)									
Oconee 1	Anderson, S. C.	269	11-67	2452	PWR (BW)	0	2.16	36	89	343	514	693		61
Oconee 2		270	11-67	2452	PWR (BW)									
Oconee 3		287	11-67	2452	PWR (BW)									
Oyster Creek	Atlantic City, N. J.	219	11-64/4-69	1600	BWR (GE)	0.198	4.84	33	136		1,153			66
Palmodes	Benton Harbor, Mich.	255	3-67	2200	PWR (GE)	0.037	4.67	26	116	189	460	570		60
Pathfinder	Sioux Falls, S. D.	130	5-60/3-64	190	BWR (AC)	0.013	8.08	56	60	67	81			
Peach Bottom 2	Lancaster, Pa.	277	1-68	3294	BWR (GE)	0.114	6.15	24	249		2,498			60-66
Peach Bottom 3		278	1-68	3294	BWR (GE)									
Pigum	Brockton, Mass.	293	8-68	1912	BWR (GE)	0.309	7.70	26	124	672	1,952	3,993		61
Point Beach 1	Manitowoc, Wis.	266	7-67	1396	PWR (W)	0.04	1.24	22	202	386				61
Point Beach 2		301	7-68	1396	PWR (W)									
Prairie Island 1	St. Paul, Minn.	182	6-68	1650	PWR (W)	0.087	3.43	16	55	244	1,272			60
Prairie Island 2		306		1650	PWR (W)									
Quad Cities 1	Clinton, Ill.	254	2-67	2300	BWR (GE)	0.06	5.37	39	256	350(25)				60
Quad Cities 2		265	2-67	2300	BWR (GE)									
Rancho Seco	Lodi, Calif.	312, 11-67		2452	PWR (BW)	0.012	0.17	4.06	303	908	1,052	1,349		61
Ravenswood	New York, N. Y.	204	4-67	2094	PWR (W)	0.417	11	27	76	220	381	615		66
Robinson 2	Columbia, S. C.	261		1584	PWR (BW)	0.105	3.74	23	49	77	185	155		67
Roseville	Hot Springs, Ark.	313, 11-67		2250	PWR (W)	0	1.18	28	399	897	2,910	5,384	7,528	67
Salem 1	Bridgetown, N. J.	272	9-68	2150	PWR (W)									
Salem 2		311	9-68	2150	PWR (W)									
San Onofre	Camp Pendleton, Calif.	202	3-67	1347	PWR (W)	0	8.83	22	97			>1,000		60
Seabrook	Portsmouth, N. H.	340	Tabled	2640	PWR (W)	0.444	37.12	80.20	293	946	2,192	3,504		48
Seola 1	Dothan, Ala.	348, 10-69		2660	PWR (W)	0.018	2.53	11.08	84	124	237	339		71
Sequoyah 1	Chattanooga, Tenn.	327, 10-68		3423	PWR (W)	0.08	6.11	25	282	382	584	684	718	60
Sequoyah 2		328, 10-68		3423	PWR (W)									
Shoreham	Stearford, Conn.	322, 5-68		1593	BWR (GE)	0.6	7.5	21	168	1,403	2,462	4,586		68
Surry 1	Newport News, Va.	280	4-68	2441	PWR (W)	0.005	0.79	46	179	308	913	1,340		66
Surry 2		281	4-68	2441	PWR (W)									
Three Mile Island 1	Harrisburg, Pa.	289	5-68	2452	PWR (BW)	0.25	27	137	630	996	1,217	1,796		67
Three Mile Island 2		320, 4-68		2468	PWR (BW)									
Trojan	Portland, Oreg.	344, 6-69		3423	PWR (W)	0.155	5.10	48.13	79	142	673	941	1,011	64
Turkey Point 3	Coral Gables, Fla.	250	4-67	2097	PWR (W)	0	0	42	232	818	1,384	1,379		66
Turkey Point 4		251	4-67	2097	PWR (W)									
Vermont Yankee	Keene, Vt.	271	12-67	1600	BWR (GE)	0.48	6.41	25	110	127(25)				60
Yankee	North Adams, Mass.	27	6-61	600	PWR (W)	0.176	3.03	29	104					50
Zion 1	Waukegan, Ill.	195	12-68	3391	PWR (W)	1.26	56	189	447	923	3,144	6,388		60-61
Zion 2		304	12-68	3391	PWR (W)									
Pickering 1	Toronto, Canada		Under construction	1742	PWR (W)	31.6(4)	206(6)	499(12)	1,407	2,424(28)				68
Pickering 2				1742	PWR (W)									
Hartlepool	Newcastle, U. K.		Site being prepared		GCR	0.140	708	508	838	2,135	1,835	3,071		
Tokai-Mura	Tokyo, Japan		Operating	585	GCR	3	68	49	652	925	2,300	2,730	4,100	61
Kahl/Main	Frankfurt, Germany		Operating	60	BWR	11.9(1.3)	190(6.2)	546(13.4)	2,923(24.9)			7,422	9,813(62)	
BASF	Mannheim, Germany		Contract being negotiated			7(1.3)	542(6.2)	1,114(12.4)	2,197(24.9)	2,055(31)	5,970(42)	7,193	9,915(61)	
Vatten	Stockholm, Sweden		Proposed			21	610	1,060	1,210	1,335				

*Docket number is preceded by 50- in all cases.

+GE = General Electric Company; W = Westinghouse Electric Corp.; CE = Combustion Engineering, Inc.; BW = Babcock & Wilcox Company; AC = Allis-Chalmers Manufacturing Company.

‡Numbers in parentheses are the distances that differ from those shown in the column headings.

§Not known.

¶With nuclear superheat.

penetration-pressurization equipment, and provisions to handle radiolysis of water and subsequent recombination of the radiolytic gases.

The Newbold Island site, for which action on a construction permit is still being awaited, is of interest to this discussion. The application for a construction permit was filed in February 1970, but, before this was done, the Public Service Electric & Gas Company of New Jersey requested that the Advisory Committee on Reactor Safeguards (ACRS) give an opinion on the site. (This was done at least once previously, in November 1962, for the City of Los Angeles reactor at Malibu, Calif.) Newbold Island is only about 6 miles from the Burlington site (discussed earlier), which was abandoned in favor of the less-sensitive Salem site (see Fig. 2). On Sept. 10, 1969, an ACRS letter¹⁵ was issued which concluded that the site for the suggested plant was "not unacceptable" with respect to the health and safety of the public.

It appears that the population distribution described initially by Indian Point, and subsequently by both Indian Point and Zion, has been used as an

"upper limit," since requests for approval of sites with larger population distributions have been either withdrawn or severely challenged. In accepting the Midland site, a small step beyond past practice was taken. If the Newbold Island site is accepted, the step will certainly be significant.

Foreign Siting

One way to focus more clearly on U. S. siting policies is to compare what is being done in other countries¹⁶ with U. S. experience. Figure 3 gives population distributions for foreign sites nearest the largest number of people and compares them with Indian Point data for 1960 and 1980.

The British have had much experience in reactor siting, and, although it has not yet been necessary their limited space and large population will force them to use urban sites if nuclear development is to continue in that country. The Ministry of Power has been working toward a formal siting policy for gas-cooled reactors with prestressed-concrete pressure vessels. The

Fig. 3 Cumulative population vs. distance for foreign plants. (The U. S. Indian Point plant is included for reference.) —, operating reactors, ----, reactors being considered or under construction.

guidelines that now appear most probable are an average population density of 4.2 persons per acre out to 20 miles. A 30° sector may have an average population density of 20 persons per acre, and this would allow siting at the edge of a rather densely populated area. Additional criteria may be established for control of activities near the plant; within $\frac{1}{2}$ mile of the plant, plans must be made to evacuate the population, and this requirement may be extended out to 2 miles; within 5 miles the Ministry of Power maintains control over new construction (see Fig. 3). Figure 3 also gives data for sites nearest the greatest number of people in the United Kingdom (Hartlepool), Japan (Tokai-Mura), Canada (Pickering), and Germany (Kahl/Main). Since the proposed BASF site was not approved, the data in Table 2 and Fig. 3 suggest that several nations with nuclear power plants are tending toward the same maximum population distributions. However, because the other countries, except Canada, are more densely populated than the United States, in a short time they may be obliged to use sites that are closer to urban areas than those in the United States.

Effect of Engineered Safety Features on Site Selection

Engineered safety features have increased in importance since the first power reactors were designed and probably will continue to do so. Early plants placed primary and rather complete reliance on the containment sphere for preventing release of fission products in the event of an accident. In more recent plants with higher power levels and greater fission-product inventories, engineered safety features have increased in both number and degree of sophistication. Plants in operation or in final stages of construction have been "backfitted" with those safety features which appear to have significant values. It may be anticipated that, as time and technology advance and as plant sizes increase, more extensive and improved engineered safety features will be included in plant designs.^{14,15,17} Such systems for a typical large plant, even when operating at partial effectiveness, have the capability of reducing the consequences of a design-basis accident to acceptable levels. As progress continues and more experience is gained, engineered safety features will probably be credited to a higher degree as barriers to fission-product release.

Plant-Site Relations. Attempts were made to determine whether correlations between sets of site and plant characteristics could be established. In pursuit of this aim, a number of plant and site

parameters considered to have a bearing on site acceptability were chosen. These are (1) thermal power, (2) age of the plant, (3) containment leakage rate, (4) distance to population center, (5) distance to metropolitan area, and (6) number of engineered safety features. Various combinations and permutations of these parameters were studied, but no meaningful relations could be found. Although on the surface it seems a little peculiar that the number of engineered safety features is not relatable to the site, the lack of correlation may be due to at least two factors. The first factor is the practice of the reactor manufacturers to sell a more-or-less standard plant, regardless of the location. This, of course, militates for the maximum number of safety systems on all plants. The rationale for this practice is that there should not be less safety because a plant is located in a remote area.

The second factor that masks the effect of engineered safety features on siting is the AEC practice of retrofitting new safety features to older plants when these features are judged to substantially improve the safety of the plant. Such features as emergency core-cooling systems have been installed in plants that were originally built without them, and, unless the history of the plant is investigated carefully, the fact that a safety feature was installed subsequent to completion of the plant may go unnoticed.

It can be concluded that "siting by equation" is not now possible; thus reliance on experience is necessary. This, of course, places a significant burden on the utility since the time required to conceive, design, construct, and place plants in operation may be 7 to 9 years or, in many cases, even longer.

New Engineering Developments. It is widely believed that the accumulation of satisfactory operating experience and improved quality assurance are all that is required to permit closer-in siting. If not these, improved containment, more conservative designs, and/or additional safety features can produce nuclear plants that are sufficiently safe for that purpose. Developments in antiseismic design, reliability concepts, and underground and/or underwater siting may also have significant effects on closer-in siting.

Reliability and Quality-Assurance Techniques. The application of engineered safety features for reducing the effects of reactor accidents and the employment of innovative siting for improved safety are, as mentioned above, important to the advancement of nuclear power, but the practice of quality assurance and better understanding of component reliability will also contribute to safer plant operation.

In pursuit of quality assurance, Appendix B, Quality Assurance Criteria for Nuclear Power Plants, of 10 CFR 50 was first published in the *Federal Register* on Apr. 17, 1969, and is now in effect as part of the regulations.² The introduction to this appendix states that quality-assurance requirements for design, construction, and operation are established by the subject document. With the use of these guidelines and with operation based on the framework already established for codes and standards, it is possible to place the design, procurement, and testing of plant components on a firm footing.

CONCLUSIONS

It appears that metropolitan siting is not a near-term option. However, it has been shown that closer-in siting is already developing in Europe because their power needs and their population density are great; and it will be many years before we in the United States must face up to this circumstance. Furthermore, increased and improved safety through experience with presently operating plants and engineered safety features, coupled with some change in public attitude toward nuclear power, will be necessary to further the trend toward closer-in siting in the United States.

In a speech, AEC Commissioner C. E. Larson stated:¹⁸

The AEC will continue to rely on a conservative approach of combining low population density requirements and engineered safety features to assure adequate protection to public health and safety. This position will be modified appropriately, based on the results of safety R&D programs and the accumulation of further operating experience.

The process of site approval now practiced by the AEC is based on sound and prudent philosophy, but this is not to say that the process could not be strengthened and speeded up. In an expanding technology, there is a natural aversion to being too explicit in the statement of requirements since this would tend to solidify the situation and dangerously slow down technological development. In the next decade, sites now available to the utilities, which are closer than presently acceptable sites, may be found acceptable by the AEC if the plants are designed to the rigorous codes and standards with sufficient quality-assurance and safety features to assure safe and reliable operation. Specific problems, such as those associated with the prevention of core melting, fission-product trapping, heat removal, containment in the event of an accident, and effluent

control under normal operating conditions, will remain under study in order to assure their acceptability. Since the demand for electricity doubles about every 10 years, the long-range siting problems are more complex. Plants will be quite large, and power-distribution networks will be more extensive and of greater capacity. The hope is that the experience by that time will be such that the public will accept nuclear power plants as safe, clean, and attractive additions to the communities that need electric power.

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2.7 AEC Working Paper on Population Density Around Nuclear Power-Plant Sites

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Abstract: This article reviews a proposed regulatory guide, "Population Distribution Around Nuclear Power Plants," which the AEC considered issuing to provide guidance on acceptable population characteristics for a nuclear reactor site. The proposed AEC guide involved cumulative population projections within 5, 20, and 40 miles of the site. Two techniques for evaluating population distributions around sites are also discussed. The first technique is the use of an unweighted cumulative population vs. distance. The second technique is called a site population factor for the purpose of comparing sites, and this technique weights population at longer distances less than the population adjacent to a site. The AEC is continuing work on these potential tools. All presently approved sites have been found to be acceptable in terms of population density as well as by other considerations.

The AEC has had a long-standing policy that encourages the siting of nuclear power plants away from densely populated areas until additional operating experience has been obtained. The policy was established in the AEC's Reactor Site Criteria that initially appeared in 1962 in the *Code of Federal Regulations*.¹ Over the years, nuclear power-plant sites have been evaluated by the AEC Regulatory Staff from a radiological-risk standpoint by comparing the results of design-basis dose computations with the numerical guidelines given in 10 CFR 100 for individual exposure. As to population risk, the AEC criteria state that "where very large cities are involved a greater distance may be necessary because of total integrated population dose considerations." However, the AEC has not provided specific guidance as to acceptable population characteristics for a nuclear power-plant site.

The AEC has considered issuing an interim regulatory guide, "Population Distribution Around Nuclear

Power Plants," to provide this guidance. As part of the process of developing this guide and other related working papers, discussions were held with a number of persons and groups, both inside and outside the nuclear industry, to obtain their views. The draft guide and related papers were disseminated in WASH-1308 (Ref. 2) and publicly released³ in April 1974. The main thrust of the proposed guide would be to provide further guidance, from the standpoint of compliance with 10 CFR 100, as to low-population-density characteristics and to require a sufficient justification for sites where the population densities are higher.

It is important to emphasize that the guide was not formally issued but only circulated for discussion. As such, it is an example of the AEC's continuing work in this area. The requirements for siting are still those set out in 10 CFR 100.

This article reviews the proposed AEC guide and discusses two techniques for evaluating population distributions around nuclear power-plant sites.

REQUIREMENTS OF PROPOSED GUIDE

The proposed AEC guide contained the following five requirements that expressed the AEC Regulatory Staff's position on acceptable population distributions around nuclear plant sites.

1. Applications for sites having a cumulative population projected from the date of application for a construction permit, as indicated in Item 2, greater than 30,000 within 5 miles, 500,000 within 20 miles or 2,000,000 within 40 miles should:

- (a) Present an analysis of alternative sites, including a showing that the proposed site offers significant ad-

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vantages from the standpoint of environmental, economic or other factors.

(b) Provide state-of-the-art engineered safety features to assure that the conservatively calculated consequences of postulated design basis accidents are significantly below the dose guidelines of 10 CFR 100.

(c) Have a minimum exclusion distance of at least 0.4 mile and a low population zone of at least 2 miles.

2. If population projections indicate that any of the values in Item 1 would be exceeded during the plant lifetime, a detailed study of economic and population growth patterns for at least 10 years after the date of application for the construction permit should be performed. The guideline values in Item 1 will be deemed to have been exceeded if (a) the detailed 10-year projection indicates that any of the guideline values are exceeded, or (b) at the time of the construction permit application, any of the guideline values can be reasonably expected to be exceeded by more than a factor of two over the projected lifetime of the plant.

3. Plant sites which fall below the population criteria of Item 1 above, and which can reasonably be expected to remain at a population level less than these guidelines over the projected plant lifetime will be individually evaluated against the dose guidelines of 10 CFR Part 100 and a detailed 10-year projection need not be performed.

4. Significant unusual population distributions within the distances specified in Item 1 above will also be taken into account in determining site acceptability.

5. Should the population at any approved site rise to unexpectedly large values during the plant lifetime, the AEC may review the population growth to determine whether additional engineered safety features should be provided or plant operations modified.

JUSTIFICATION FOR THE GUIDE

Population distributions around a proposed nuclear power facility are considered primarily to assure that risks to the population as a result of any accidental releases of radioactivity are adequately taken into account in the selection of the site. Since the routine releases of radioactivity allowed under AEC regulations are extremely small, the risks to the population from these releases, even assuming a very high population density around the facility, can be fairly characterized as negligible.

Under 10 CFR 10, applications are required to postulate a large release of fission products within the reactor containment and, on the basis of acceptable assumptions for various design-basis accidents provided by the AEC Regulatory Staff, to calculate off-site exposures from such an assumed release. These calculated exposures may not exceed exposure guidelines established in 10 CFR 100, and, owing to the resulting conservative plant design, the risk to any individual from accidents is small. Part 100 also specifies that

special considerations may apply when large population centers are near proposed power-reactor sites because the large number of persons who potentially could be exposed to radioactivity in the event of an accident does present some small additional increment of risk to the total population. The use of the proposed AEC guide should assure that, with reactors of current technology, sites that have high population densities are not used in the absence of sufficient justification.

The accidents postulated for the radiological calculations require a whole sequence of failures of safety equipment. Quantitative work on the likelihood of these failures occurring has been performed in the AEC's Reactor Safety Study headed by Prof. Norman Rasmussen of the Massachusetts Institute of Technology. A draft report⁴ on the work, released for comment in August 1974, shows that realistic probabilities of these events are extremely small.

The AEC recognizes that siting of reactors near highly populated centers presents some additional increment of risk over that presented by a low-population-density site. Therefore the intent of the guide is to encourage siting in low-population-density areas unless high-density areas offer significant environmental, economic, or other advantages.

Over 90% of the sites proposed to date would meet at least two of the population levels indicated in item 1 of the proposed AEC guide. About 70% of the sites proposed to date have current populations that are below all levels indicated in item 1.

POPULATION DISTRIBUTION AROUND APPROVED AND PROPOSED REACTOR SITES

As background information the AEC, along with the proposed guide, presented data on two techniques for evaluating population distributions around reactor sites. The first technique is the use of an unweighted cumulative population vs. distance. The second technique is called a site population factor for the purpose of comparing sites, and this technique weights population at distances further from the site less than the population adjacent to a site. The latter concept has played a key role in the AEC Regulatory Staff's analysis of population patterns and in its selections of the cumulative population values given in item 1 of the proposed guide.

Cumulative Population Data

Population distributions for a representative sample of existing sites are shown in Figs. 1 and 2 as a

Fig. 1 Typical site population distribution (0 to 5 miles).

Fig. 2 Typical site population distribution (5 to 50 miles).

function of distance in relation to hypothetical cumulative population curves that correspond to uniform population distributions of 250, 400, and 1000 people per square mile based on 1970 census data. The cumulative populations are plotted out to 5 miles in Fig. 1 and out to 50 miles in Fig. 2. The same type of information is presented for relatively high-population-density sites in Figs. 3 and 4.

The present informal AEC Regulatory Staff population criteria for siting nuclear plants are based on a case-by-case comparison against existing high-population sites. Indian Point and Zion, which have received construction permits, are the sites with the highest population densities. The AEC Regulatory Staff has issued a safety evaluation for Limerick, which also has a high-population-density site, and Public Service Electric & Gas Company of New Jersey had proposed to construct two power reactors at Newbold Island, another high-density site. However, in October 1973 the AEC informed the utility that, on the basis of its environmental review, it had concluded that the two reactors should be moved from Newbold Island, south of Trenton, N. J., to the Artificial Island site in Salem County, N. J. The principal reason for the AEC's

decision was the fact that the population density at the Newbold site is significantly larger than that at Salem. Both the Ravenswood site of Consolidated Edison Company and the Burlington site of Public Service Electric & Gas Company of New Jersey were dropped before formal consideration by the AEC Regulatory Staff.

A comparison of six high-population-density sites to the maximum 1980 population level that has been approved at the construction-permit stage is presented in Table 1. The lower population levels from the proposed AEC guide are given for reference. These are the levels above which state-of-the-art engineered safety features and an analysis of alternate sites would be required.

Site Population Factor

The AEC Regulatory Staff has developed another index for use in comparing population distributions around reactor sites which it refers to as the site population factor (SPF). The index weights each discrete element of population by a decreasing function of its distance from the site and can be used to numerically rank any population distribution in rela-

Fig. 3 High-population-density site distribution (0 to 5 miles).

Fig. 4 High-population-density site distribution (5 to 50 miles).

Table 1 Cumulative Population (1980 Projections)

	5 miles from site	20 miles from site	40 miles from site
Maximum population approved at construction-permit stage	110,000	1,200,000	13,000,000
Proposed lower population definition (400 people/sq mile)	30,000	500,000	2,000,000
1000 people/sq mile	78,530	1,256,000	5,026,000
Zion	106,615	846,515	5,184,515
Indian Point	70,053	1,179,611	12,882,240
Newbold Island	122,335	1,843,935	7,097,935
Burlington	236,000	5,150,000	8,800,000
Ravenswood (1970)	3,599,248	12,779,748	18,324,030
Limerick	93,262	960,800	6,958,800

tion to a hypothetical site. For comparison purposes the AEC has chosen to normalize the total weighted population within each selected bounding radius to a hypothetical site with a uniform population distribution of 1000 people per square mile. The weighting factor used is the distance from the site raised to the negative 1.5 power. This factor was selected because it follows very closely the increase in atmospheric dispersion and dilution with distances that are used in analyses of assumed releases of radioactivity from a reactor.

The AEC has calculated SPF's for existing and proposed power-reactor sites at annular distances of 1, 2, 3, 4, 5, 10, 20, 30, 40, and 50 miles. The SPF's calculated at each of the distances are then normalized to the hypothetical population density of 1000 people per square mile. Therefore a site having an SPF of 0.4 at some selected bounding radius is equivalent to a hypothetical site that has 400 people per square mile uniformly distributed out to that distance.

Table 2 presents the maximum SPF and the SPF at 50 miles that the AEC Regulatory Staff has calculated

Table 2 SPF's for U. S. Nuclear Reactor Sites

Plant	Maximum SPF	SPF at 50 miles	Plant	Maximum SPF	SPF at 50 miles
Ravenswood	45.0	15.5	Calvert Cliffs	0.188	0.142
Burlington	2.84	0.304	Maine Yankee	0.184	0.128
Indian Point	1.49	1.49	Turkey Point	0.182	0.172
Newbold Island	1.35	1.19	Susquehanna	0.180	0.175
Zion	1.02	0.730	La Salle	0.163	0.077
Standard (1000/sq mile)	1.00	1.00	Vermont	0.161	0.122
Limerick	0.903	0.835	Surry	0.159	0.159
Shoreham	0.657	0.451	Sequoyah	0.153	0.112
General Electric			Rancho Seco	0.152	0.135
Test Reactor	0.638	0.583	Prairie Island	0.151	0.151
Bailly	0.619	0.616			
Midland	0.557	0.238	Duane Arnold	0.144	0.097
Fermi	0.531	0.50	Monticello	0.143	0.143
H. B. Robinson	0.514	0.095	Yankee Rowe	0.135	0.117
Oyster Creek	0.490	0.238	Quad Cities	0.118	0.091
Tortuguera	0.481	0.38	Palisades	0.109	0.099
Haddam Neck	0.451	0.370			
Millstone	0.435	0.300	Oconee	0.093	0.088
Beaver Valley	0.432	0.412	FitzPatrick	0.093	0.082
Three Mile Island	0.390	0.304	Hutchinson Island	0.091	0.057
Waterford	0.388	0.215	North Anna	0.085	0.061
Peach Bottom	0.387	0.308	Browns Ferry	0.076	0.076
Salem	0.385	0.353			
San Onofre	0.343	0.240	Atlantic	0.072	0.072
Portland	0.339	0.339	Watts Bar	0.063	0.053
Pilgrim	0.322	0.322	Point Beach	0.061	0.061
			Arkansas	0.054	0.032
Babcock & Wilcox			Kewaunee	0.055	0.055
Test Reactor	0.295	0.133			
Humboldt Bay	0.274	0.090	La Crosse	0.040	0.038
Santon	0.273	0.099	SI FOR	0.038	0.036
Cook	0.273	0.144	Farley	0.038	0.037
Fort St. Vrain	0.256	0.224	Cooper	0.036	0.018
			Big Rock Point	0.034	0.021
Ginna	0.247	0.180			
Davis Besse	0.234	0.147	Brunswick	0.034	0.027
Trojan	0.229	0.192	Diablo	0.030	0.026
Zimmer	0.218	0.194	Crystal River	0.030	0.020
McGuire	0.218	0.186	Hatch	0.020	0.018

for both approved and proposed sites. The maximum SPF's are listed in descending order of magnitude.

In another examination of SPF's, the AEC Regulatory Staff was able to characterize those for existing sites into five representative groups. These SPF's are shown in Fig. 5 with a typical reactor from each group.

Fig. 5 SPF as a function of distance.

Group 1 consists of sites that are fairly close to metropolitan areas having SPF's of 1.00 or greater. Group 2 includes sites in which a large city is located 10 to 30 miles away; the maximum SPF's are less than 0.3. Sites in group 3 are in relatively unpopulated areas but have a small town located 1 to 5 miles from the reactor. In this case the SPF's usually peak in the 1- to 4-mile region and then fall to a constant level of about 0.3 or less. Group 4 contains sites in unpopulated areas whose SPF's remain constant at fairly low levels. Group 5, sites that have large metropolitan areas located 40 or more miles from the reactor, have SPF's which are low but of increasing function.

The SPF's were analyzed for the various reactors at 5, 20, and 40 miles from each site, the distances used

to specify maximum allowable cumulative populations in the proposed AEC guide. About 90% of the currently approved sites have SPF's below 0.4 at 5 miles and also at 20 miles. About 85% of the sites have SPF's below 0.4 at 40 miles. An equivalent uniform cumulative population based on a density of 400 people per square mile was chosen at each of these distances as the definition of a low-population site in the proposed guide.

When the sites are examined only from the standpoint of cumulative population, about 70% would fit the cumulative population definition of a low-population site at all three distances. However, the cumulative population values do not take into account maldistributions of population that could occur between the specified distances of 5, 20, and 40 miles. For this purpose the SPF technique provides a better measure of relative risk. Thus this concept could aid the AEC Regulatory Staff in determining the acceptability of a plant meeting the cumulative population criterion but having most of its population just outside the 5- or 10-mile distance. The AEC Regulatory Staff expects to further develop the SPF concept.

The AEC Advisory Committee on Reactor Safeguards (ACRS) has also done some work on developing a quantitative basis for comparing nuclear power-plant sites. In August 1974 the ACRS released minutes of its committee and subcommittee meetings that discussed a site population index developed for this purpose.⁵

CONCLUSIONS

The AEC is continuing work on generally applicable siting guides for nuclear power plants and on the technique for weighting SPF's as a function of distance. Population density has always been an important consideration in the process of determining the suitability of any proposed site and nuclear plant design.

The proposed AEC guide and the population weighting technique discussed in the working paper represent potential tools for evaluating sites from a population-distribution standpoint. However, the requirements of 10 CFR 100 still have to be met for a nuclear plant independent of the guide or other techniques that are under consideration.

All presently approved sites have been found, in terms of population density as well as other considerations, to be acceptable by the AEC. It is the policy of the AEC to encourage the siting of nuclear power plants at some distance from densely populated areas until additional operating experience has been obtained.

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3. RISK CRITERIA

3.1 Reactor Safety and Siting: A Proposed Risk Criterion

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[*Nucl. Safety*, 8(6): 539-548 (November-December 1967)]

Editor's Note: The following article is taken from papers presented earlier this year by Mr. Farmer at the IAEA Siting Conference in Vienna in April and the ANS Meeting in San Diego in June. It is interesting to note the differences in U. K. siting practices, as indicated herein, from those specified in Title 10, *Code of Federal Regulations*, Part 100, for U. S. reactors.

A paper by J. R. Beattie on Risks to the Population and the Individual from Iodine Release, was referenced in, and appended to, Mr. Farmer's paper. Because of its general interest, and its close association with the following article, Mr. Beattie's paper is included in Sec. V of this issue of *Nuclear Safety*, page 573.

Abstract: The building of a reactor implies the acceptance of some finite degree of risk. No engineering plant and no structure is entirely risk free, and there is no logical way of differentiating between credible and incredible accidents. The incredible is often made up of a combination of very ordinary events—for example, the breakdown or deterioration that occurs in normal plants and their measuring instruments—and the credible may be exceedingly improbable. The logical way of dealing with this situation is to seek to assess the whole spectrum of risks in a quantity-related manner and thus provide for the choice of sites for nuclear power stations.

The sites selected in the past for nuclear power stations reflected the best available judgment of the balance between the safety of a particular reactor and the size of the population theoretically at risk. This was good enough to get nuclear power programs started. With relatively few stations there was a fair degree of freedom in choosing where they had to be. By now, however, many countries, some of which have high population densities, have accepted the major role of nuclear power. This implies the need to site nuclear power stations strategically in rela-

tion to the demand for electricity if their full economies are to be realized. In order, therefore, to derive full advantage from the development of nuclear power, it is now necessary to identify and adopt quantitative safety standards to which siting considerations can be related.

The growth of nuclear power in the next decade will be based on the few reactor types that are now well established. Decisions being taken in many countries in relation to their design and siting will be far reaching and will apply to some tens of installations, each of which will continue to operate for perhaps 30 years or more. It is the safety of these reactors that is now under review.

The very substantial effort made in the United Kingdom and other countries to improve the economic performance of reactors has been accompanied by a corresponding effort in the study of a wide range of potential accidents in order to estimate the quantity of fission products that might be released. The obsessive preoccupation with which these studies have been pursued has diverted attention from two fundamentally important considerations, namely, the likely frequency of accidents and the consequent risks to people.

It is generally agreed that the isotopes of iodine, and particularly ¹³¹I, carry a greater threat to health than any of the other fission products that might be released in a reactor accident, and the International Commission on Radiological Protection (ICRP) has published findings of the risk of developing thyroid cancer through the inhalation of radioactive iodine.¹ If a way could be found to estimate the likelihood of releasing various quantities of ¹³¹I from a particular reactor, the risk associated with its operation on any site could be calculated.

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The building of a reactor implies the acceptance of some finite degree of risk. No engineering plant and no structure is entirely risk free, and there is no logical way of differentiating between credible and incredible accidents. The incredible is often made up of a combination of very ordinary events—for example, the breakdown or deterioration that occurs in normal plants and their measuring instruments—and the credible may actually be exceedingly improbable. The logical way of dealing with this situation is to seek to assess the whole spectrum of risks in a quantity-related manner, and the purpose of this paper is to show how this may affect the choice of sites for nuclear power stations.

Introduction to the Probability Approach

It may be helpful in advance to set out in sequence the main steps on which the analysis is based and to deal later in more detail with the analysis itself and with examples of its application. Some assumption of plant failure or maloperation provides the starting point for a safety analysis. A measure of risk can be obtained by estimating the probability of the failure and assessing the consequences. Any initiating event—for example, failure of piping, delays in the operation of control systems, loss of circulator power, or combinations of these—can set up an accident sequence that can follow many paths. The likelihood of following any one path depends on the performance of many items of plant, and, from knowledge of this performance, a probability can be assigned to each pathway. The full safety evaluation then comprises a spectrum of events with associated probabilities and associated consequences. A probability vs. consequence diagram can then be drawn, as in Fig. 1, with the average time interval between

Fig. 1 Probability vs. consequence diagram.

events (i.e., reactor-years) as a convenient scale for probabilities, and, for consequences, the equivalent ground-level release of ^{131}I .

In a similar plot in Fig. 2, area A is one of low risk, and area B one of high risk; all parallel lines of equal slope -1 join points of equal risk in terms of curies released per year. One such line might be used as a safety criterion by defining an upper boundary of permissible probability for all fault consequences.

Fig. 2 Possible means for establishing an upper boundary of permissible probability for all fault consequences.

An Example of the Probability Approach

This concept of the probability-associated risk is illustrated by a typical fault study on an advanced gas-cooled reactor (AGR). The accident chosen is a failure in the high-pressure coolant system which leads to depressurization of the reactor. The AGR pressure vessel is constructed of prestressed concrete² through which a number of penetrations provide access for refueling, for maintenance of gas circulators, for steam and water connections, and for gas repurification. Many openings are about 12 in. in diameter, but a few are larger, and in the following example a hole 24 in. in diameter is taken. The gas pressure drops steadily and approaches atmospheric pressure in about 5 min after the failure. The reactor is shut down at the onset of the fault. Gas circulators may continue to run for some time until the main steam turbine is tripped by low steam pressure, and some circulators may be driven by an external power supply and remain unaffected by the fault. Alternative power from diesels or gas turbines is available for automatic or manual connection.

The likely result of the accident is a progressive cooling of the reactor fuel and moderator, with no

release of fission products. In the analysis of the accident, the required data on gas-discharge rates, circuit pressures, and heat-transfer phenomena are all well established. The uncertainties lie in the behavior of the plant and are entirely in the domain of normal engineering experience. This leads to consideration of delays and partial failure of the shutdown system and to the loss of power to groups of circulators (depending on their connections) and delays in commissioning auxiliary power supplies. These are some of the factors which could lead progressively to a worsening of the accident through diminishing effectiveness of coolant flow and a subsequent increase in fuel temperature.

credited with two values: one for the likely frequency of its occurrence and the other for the quantity of ¹³¹I that might be released. This is illustrated in Fig. 5, in which the failure of a 24-in. penetration is taken as the initiating event.

It is clear from the foregoing analysis that the overall probability of arriving at a given end point depends on:

1. The reliability with which the protective system can shut down the reactor
2. The reliability of main and auxiliary power supplies to maintain coolant flow
3. The probability of a structural failure as an initiating event

Fig. 3 Alternative sequences following initial fault condition. Numbers are given to identify the temperature transients in Fig. 4.

A systematic representation of the alternative routes the accident might follow is shown in Fig. 3. Each junction indicates a possible variation of performance of the plant, instruments, or operator. At each junction the probability of proceeding along alternative paths is assessed on the basis of experience with identical or similar equipment.

It has been the practice to select a few accidents combining varying degrees of pessimism in the choice of delay times for the shutdown system; and in the availability of auxiliary power and to explore the consequent temperature history of the fuel. A typical set of curves is presented in Fig. 4. The presentation is then used in various arguments concerning the likely, unlikely, or maximum credible event, with the choice being entirely a matter of opinion.

As a result of such a study, it is possible to identify a series of end points, each of which can be

Fig. 4 Possible temperature transients due to 24-in. circuit breach. Numbers follow the sequences given in Fig. 3.

POINT	INITIATING EVENT	ASSUMED PROBABILITY OF INITIATING EVENT, reactor-years
C AND A	RAPID TRIP, 2 BLOWERS CONTINUE	0.99×10^{-6}
B	SLOW TRIP, 2 BLOWERS CONTINUE	0.01×10^{-6}
D	STATION BLOWERS FAIL	0.01×10^{-6}
E	NO TRIP	
1 AND 2	FEWER THAN 4 BLOWERS AVAILABLE AT 4 min	0.1×10^{-6}
3	6 BLOWERS AT 4 min	0.2×10^{-6}
4	8 (ALL) BLOWERS AT 4 min	0.7×10^{-6}

POINT REFERENCE	TRANSIENT CURVE IN FIG. 4
C4 A4	1257
C3 A3	1258
C1 C2	1259
B4	1357
B3	1358
B1 B2	1359

Fig. 5 Possible consequences associated with depressurization through 24-in. breach. C indicates a rapid trip from two trip parameters. A indicates a rapid trip from one trip parameter, the other having failed; the chance of this is put at 10^{-2} . C3 and A3 both involve, at their respective probabilities, a few of the koller cans releasing fission products, assessed at 10^2 curies. C4 should apparently involve no release, but for the nominal condition of C4 (e.g., nominal can temperature starting at 800°C) there is a chance that all reactor temperatures could be 20°C above normal and that the can bursting data could be the equivalent of 20° under-estimated. Together these factors could cause the release of 10^2 curies for A3. However, the probability of each is assessed at 10^{-2} . Thus C4 is a "tail of distribution" similar to B1, but the nominal release for C4 is zero.

Reliability of the Protective System

The protective system receives information from the reactor and its associated equipment (blowers, pumps, boilers, etc.), compares this information with preset limits, and initiates alarms or actions to

shut down the reactor and stop or start pumps, blowers, etc. A typical system is shown in block-diagram form in Fig. 6 and in more detail in Fig. 7. The required performance of the complete system can be resolved progressively into a requirement for each subsection. A typical subsection shown in Fig. 8 converts signals from pressure and temperature measurements into an action that releases control rods.

Fig. 6 Simplified control and protection system.

In order to provide the protection specified, attention is finally directed to the ability of individual pieces of equipment to function as required. This equipment will consist of resistors, capacitors, transistors, small hydraulic and pneumatic devices, etc., and the performance of the assembly can be derived from relevant experience of the whole or its parts. The reliability of control equipment has been studied for many applications in the last few years, and techniques that permit prediction from basic components have shown good correlation with service-life experience (Tables 1 and 2).

A protective system can be engineered to meet very high performance requirements, so a reasonable estimate can be made of its potential rate of failure, and, through progressive experience and standardization, very low failure rates can be achieved. Some control systems are required to respond correctly on a very large number of operations, as for example, the automatic landing of aircraft; in this case a correspondingly low probability of failure is required (1 in 10^7 landings). In reactor safety systems, critical demands are made relatively infrequently, and some reactors may be adequately protected by equipment, having a probability of failure in the range of 1 in 10^4 to 1 in 10^5 per demand.

Reliability of Power Supplies

The provision of power supplies for main circulators, water supplies to reactor or boilers, etc., does not introduce any new concept, and the same equipment can be used in nuclear plants as in fossil-fueled

Fig. 7 Reactor protective system.

Fig. 8 Diagram of typical reactor safety circuit.

plants (i.e., transformers, switchgear, and standby power units). Main and auxiliary supplies are closely interconnected, and Fig. 9 shows a typical arrangement.* In such a system in which several alternative supplies provide a reasonable degree of redundancy, the reliability required of individual components is not excessively high and is within the capacity of equipment in everyday use. Experience in the United Kingdom with similar equipment already provides a basis for the estimation of the reliability of the power system as a whole, and these estimates will improve during the working life of current nuclear and conventional plants.

*This is for illustration only and may not represent the exact arrangement for any one power station. It is not implied that this degree of interconnection and reserve power is a necessary part of any one station.

Table 1 COMPONENT FAILURE RATES

Equipment	Failure rate (% in 1000 hr)
High-stability carbon-film resistor	9.1
Carbon-track resistor (variable)	0.2
Polystyrene capacitor (fixed)	0.01
Diode valves	0.9
Alloy-germanium transistors	0.1
Silicon planar diodes	0.002

Table 2 EQUIPMENT FAILURE RATES

Equipment	Faults per year	
	Predicted rate	Observed rate
Temperature trip amplifier	2.8	2.3
Pulse-rate-measuring channel	17.5	17.4
Differential-pressure transmitter	0.45	0.76
Gamma monitor		
Dangerous failure	0.16	0.16
Spurious alarm	0.07	0.02
Other categories	0.15	0.16
Total	0.36	0.34

Reliability of Structures

Reactor structures comprise pressure vessels in steel or concrete with internal and external pipes and fittings and other structural features separating and directing the flow of coolant. The techniques used in their construction and inspection are common to a wide range of industrial plants or have evolved from conventional engineering practice. Experience in the use of pressure vessels and piping is very large, and many tens of thousands of such items of plant are in use in many countries. A review of this experience has been undertaken in the United States,¹ the United

Fig. 9 Diagram of typical interconnection of main and auxiliary power supplies.

Kingdom, and elsewhere.⁴ Although nuclear power may require an extension of traditional skills, it would be imprudent to deny the value of experience in attempting to evaluate the behavior of present and future plants. The assessment of the acceptability of a reactor necessarily includes a judgment on the likelihood of plant failure. If evaluation is limited to a judgment of the reactor as a whole, this restricts the use of skill and expertise to that available from nuclear specialists, whereas a far wider group of people have experience in structures and are competent to pass judgment on them.

Recently engineers in the U. K. Atomic Energy Authority and the Central Electricity Generating Board made an attempt to quantify opinions concerning the likelihood of failure of particular plant items. In many cases the opinions, expressed as probabilities, agreed within 1 order of magnitude. Judgments of this kind can be extended to take account of national and international industrial experience interpreted by skilled engineers in many countries.

The consequences of most faults on a reactor system, whether structural or otherwise, are limited by the diversification of protective systems and the provision of redundant equipment. Many structural faults can be accommodated by such defense, in depth, through auxiliary power units, the use of double enclosures or pipes, the subdivision of the plant, or the use of throttling devices to limit the consequences of failure. In such cases the failure rate that can be tolerated may well lie within the range predicted on the basis of experience.

There are very few cases of structural failure that cannot be accommodated in this way, as for example, the gross failure of the pressure vessel. In this case, the consequences are potentially severe; secondary protection may not be adequate, and an extremely low failure rate is required. It is not possible by extrapolation from past experience to establish the validity of a failure probability as low as 1 in 10^5 to 1 in 10^6 in the lifetime of a single unit. Such a low assessment can only be based on extensive experience of the behavior of a wide range of related types of equipment, together with experienced judgment of the effectiveness of the various means of reducing or eliminating failures through special effort in design, fabrication, and inspection and the development of lifetime testing procedures.

An alternative route available in the case of the concrete pressure vessel is to reestablish the advantage of multiple protection through a subdivision of the stressed members into thousands of cables. The validity of assuming a low failure probability then depends on a systematic appreciation of the potential modes of failure in order to support the premise that no failure routes can take place without prior warning revealed by known inspection techniques.

A Boundary Line as a Criterion

In the introductory section of this paper, it was suggested that a useful standard for a reactor might be provided by a boundary line of the type indicated as CD on Fig. 2. Although this line joins points of equal risk in curies per year, it may not represent an equal risk of casualties.* Furthermore, it is likely that most people would apply a relatively heavier penalty against the possibility of a large release than a small release. This would lead to a boundary line of greater negative slope. The one now chosen has a slope of -1.5 to reduce by 3 orders of magnitude the frequency of an event whose severity increases by 2 orders of magnitude.

The position of the line will reflect three main considerations:

1. Possible public reaction after an accident
2. The estimated number of casualties likely to arise in the population affected by the release
3. The increased risk incurred by any individual

It is necessary to define a population distribution in order to establish the collective risk 2 or individual risk 3. In view of the growing need for flexibility in reactor siting, the population chosen for the present study is typical of that around suitable industrial sites in or near large cities in the United Kingdom. Details are presented in Beattie's article (page 573) from which a standard site is chosen having a uniform population density of about 13,000 per square mile in all directions from about 0.5 mile to 10 miles from the site. This site is thus surrounded by 4,000,000 people.

The main consequence of a ground-level release of a few thousand curies of ^{131}I on the site defined above is considered in order to assess the position of a suitable limit line, as might be determined by a prediction of public reaction. It is shown later that this is unlikely to cause many casualties and, in real terms, is not an event of unusual danger when compared with other industrial activities. Nevertheless, it is unlikely that any operator or authority would willingly anticipate this event once within the life history of each plant, that is, once in 30 years.

However, the programs of reactor installation in several countries will accumulate a thousand or so reactor-years of operation within this century. Is the risk of one event within this period acceptable? The choice must surely lie between 0.1 and 10 events, and within this range a reasonable starting point is the choice of a time interval of 10^3 reactor-years, which, for a normal distribution, implies a 1-in-3 chance of a single event occurring within the period and a 1-in-3 chance that the event will not occur. The curve shown in Fig. 10 is now determined by the

*A casualty in this exercise and throughout this paper means one case of thyroid carcinoma.

Fig. 10 Proposed release criterion.

choice of this interval of 10^3 reactor-years for a release of a few thousand curies and by the slope of -1.5 . There is a transition in the slope in the range 10 to 1000 curies which serves to minimize the frequency of small releases, largely on the basis of their nuisance value.

Assessment of Risks to People

In Beattie's article (page 573), the ICRP findings¹ on the risk from the inhalation of radioiodine are used to calculate the number of cases of thyroid carcinoma that might eventually result from the release of specific quantities of ^{131}I at groundlevel. The assumptions made are principally:

1. The ICRP statement¹ that "1 rad will produce a total of 10-20 cases per million persons." A mean of 15 has been used in calculations.
2. The population density around the site extends from 0.5 mile to 10 miles radius at 12,800 per square mile.
3. The weather condition is a weighted mean of Classes A to F, as described by Pasquill,⁵ with a dispersion over a 30° sector.

The number of casualties that might arise in a population of 4,000,000 people in a period of 10 years or so after a release of 10^6 curies is calculated to be 33. The boundary line limits the frequency of this size of event to 0.66×10^{-4} per reactor-year (Fig. 10), from which the derived casualty rate is 0.002 per reactor-year. The figure assessed on the same basis for a release of 10^3 curies is 0.003 per reactor-year. Table 3 of Beattie's article sets out the corresponding values for higher releases.

The risk assessment of a reactor does not culminate in one single point but in many. A large propor-

tion will lie well within the safe zone—by several orders of magnitude in probability or severity—and thus make little contribution to the total risk. A few points may lie near the boundary line, and a summation of their associated casualty-risk rates provides the basis for the overall risk assessment. On the assumption that few points lie near the line, each carrying a casualty risk of 0.001 to 0.003 per year, it follows that the aggregate risk of casualties arising from the operation of a reactor that meets the proposed criteria on an urban site is of the order of 0.01 per year. This risk is shared by the surrounding population of 4,000,000, and, if the numbers at risk are lower, the risk rate is also lower. It is interesting to compare the nuclear risk with risk of casualties likely to arise in the same population through natural causes. According to the statistics published by the U. K. Registrar General, the annual death rate in England in a population of this size would include:

Cause of death	Annual rate
Thyroid cancer	70
Leukemia	180
Accidents	1000

The highest individual risk assessed by Beattie is 1×10^{-7} per year. This is 1 to 2 orders of magnitude lower than the risk of naturally occurring thyroid cancer⁶ or leukemia and nearly 10,000 times lower than the risk of accidental death.⁷

Considering either the individual or the community, it is obvious that the risk is very small indeed, and it is likely that a more relaxed criterion could be chosen if these risks were the only consideration.

Discussion and Conclusions

The ideas presented in this paper are not new, they have been discussed and used in part in various countries. In 1955 in Geneva, at the First International Conference on the Peaceful Uses of Atomic Energy, Marley and Fry⁸ stated, "The preceding discussion gives no indication of the probability of an actual release of radioactivity from an atomic energy installation, but this is just as important as an assessment of the effects." The adoption of this type of safety analysis has been delayed by lack of information, lack of effort, and perhaps by lack of imagination. It may be thought that the mathematical terminology and form of presentation require exactitude in analysis. This is not so; informed opinion can be expressed in the terms required for this analysis with greater meaning than is conveyed by the safety language now used.

The boundary-line concept is a simple way of defining a criterion that can be used by designers and

assessors. It does not, in itself, define a total limiting risk, since this requires the summation of all risks that may be presented in different ways by the multiple subdivision of the accident studies. In principle these can be summed, but, in my opinion, the practice of risk assessment should be allowed to develop without excessive complexity and within the simplest possible target.

Some conclusions of a general nature, but of fundamental importance to reactor safety and siting, stem from the general theses:

1. There is no need to obtain greater precision in the estimation of fission-product release than the precision that can be attributed to the probability of a release.

2. If any single feature of a reactor is used to reduce a potential release of fission products by many orders of magnitude (as by filtration, containment, or reactor control), the reliability of this feature should be established to at least the same order of magnitude. Figure 11 illustrates this.

Fig. 11 Releases as a function of protective-system reliability.

An initial fault (of chosen frequency of 10^3 reactor-years) might, in the absence of a suitable protective system, give rise to a release shown at A. The correct functioning of the protective system would reduce the release to point B. Some deterioration or failure of the equipment could lead to a larger release that would eventually reach the magnitude of A but at a lower frequency. This may be shown as C or D, depending on the reliability of the equipment. Many accident sequences would link BC or BD. It is clear that the points near C are of greater safety significance than those near B. Safety studies should

evaluate this area with great care rather than concentrate on the detail of B, as is so often the case in siting submissions. It follows then that, if protective equipment is used to modify accident consequences from A to B (i.e., by 3 orders of magnitude), its failure rate should be established as less than 1 in 10^4 in order to generate the line BD.

3. The final presentation of reactor risk is inexact and may, at best, be claimed to represent the reactor within a factor of 10. This is not an outcome of the present method, but a consequence of shortcomings in knowledge of plant performance, equally, or more so, than any shortcomings in knowledge of reactor and fuel behavior.

It is now possible to use the present form of assessment and boundary line to establish criteria for reactor sites. Should the boundary line be varied to take account of population density?

1. Those sites in the United Kingdom which might be used in a power network, as distinct from truly remote sites, vary by a factor of 10 at most in population density.

2. The estimate of risk associated with a reactor carries an uncertainty of about 1 order of magnitude. Similarly, the position of the boundary line may be placed by informed opinion also within 1 order of magnitude.

Hence reactors cannot be matched to sites with any precision. If reactors can only be graded within a factor of 10, is there any merit in a more refined subdivision of sites? There is obviously a development period in all countries permitting a growth of reactor technology and skills in operation and inspection—a maturing of siting philosophies and of public opinion. In this phase, site selection is very much a matter of political wisdom.

It is already clear in the United Kingdom that reactors must attain a standard of safety that will permit complete freedom in siting, and a similar situation will be reached shortly in a number of countries. When this goal is reached, site categorization will disappear and all reactors will need to meet a single high standard if they are to play a significant part in a power network. Less-populated sites might be reserved for the development of new types.

This paper has proposed a means by which an appropriate safety standard may be defined by any country and a means for assessing the ability of a reactor to meet the standard.

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3.2 The Development of Reactor Siting Criteria Based upon Risk Probabilities

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Abstract: Examination of the originally proposed Farmer limit-line criterion for nuclear plants led to a total individual mortality risk that is fairly high with respect to other recorded risks. For example, in the case of an equal wind-direction probability of $\frac{1}{12}$ (any 30° vector over a 360° circle), the total risk from an accidental release at the plant was only 3.4×10^{-6} but was still 34 times that caused by lightning. A new location for the limit line was therefore deduced that would satisfy both Farmer's probabilistic safety-analysis philosophy and at the same time limit the added risk to the individuals living at the exclusion radius to a very low value. Finally, the probabilistic safety analyses of three existing nuclear plants are compared with the original and modified limit lines. Hence one can observe a correspondence between calculated plant release data and the limit-line concept. We thus conclude that an integrated safety analysis based on the limit-line concept could provide the nuclear industry with an adequate reactor siting technique.

Analysis of the safety of nuclear plants has concentrated on the detailed study of a wide range of potential accidents and the subsequent release of fission products into the atmosphere. The frequency of occurrence of such accidents and the consequent risk incurred by the population living near the plant have not been considered to any extent. Farmer,¹ of the United Kingdom, proposed a criterion specifying upper limits for the release of ^{131}I in accident situations as related to their frequency of occurrence. This article

examines the application of Farmer's limit-line criterion. Total individual mortality risk per year for persons living at the exclusion distance of the plant was calculated, and this risk was then compared with present socially recorded risks. As a result, we suggest a modified limit line leading to a calculated total risk that lies within present acceptable limits. Also, the consequences of accidents calculated for a number of existing nuclear plants with the original and the modified limit line are compared.

In Farmer's criterion, only ^{131}I was considered because of its greater threat to health than other fission products that could be released in a reactor accident. Ergen² noted a radioactive dose ratio between ^{131}I and the noble gases of about 23 to 1, and Beattie³ stated that the iodine hazard is much more significant than that due to ^{137}Cs and ^{90}Sr , the isotopes next in importance. Following the risk ideas of Farmer, Beattie⁴ evaluated the risk to persons living near a nuclear plant that had a release of 10^4 Ci of ^{131}I . Alternatively, work by Otway and Erdmann⁵ resulted in the development of a method for calculating individual mortality risk from a nuclear plant.

The present work results in a modification (relocation) of the original Farmer limit line by a synthesis of

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Farmer's release-probability approach and the present socially accepted and recorded risk concept.⁵ A detailed account of the Farmer limit-line concept is given in the following section. Accidental-death statistics in the United States are then reviewed in order to suggest a socially acceptable individual risk; actual reactor accident release-probability calculations are compared with the original limit line and modified versions; and, finally, conclusions and recommendations are made to modify the original limit-line idea so that a synthesis of the two risk concepts is effected.

THE LIMIT-LINE CONCEPT

In 1967 Farmer¹ suggested a new approach for evaluating the safety of nuclear power plants which could be used as a guide for siting plants near load centers. The proposed criterion specified upper limits for the release of ^{131}I in accident situations of a given frequency. Two such curves are shown in Fig. 1.

Fig. 1 Farmer limit line.

Curve *a* has a slope of -1.0 , indicating that it joins points of equal risk in curies per year. To accommodate for expected greater public reaction to a larger release than to a smaller release, Farmer suggested curve *b*, with a slope of -1.5 . This would indicate that the frequency of an accident is reduced by three orders of magnitude when its severity increases by two orders. The second parameter defining the curve is the choice of a probability of 10^{-3} for a release of 1000 Ci. Finally, a lesser slope between 1000 and 10 Ci is used to minimize the frequency of small releases. Thus no release, no matter how small, should occur at a larger probability than 10^{-2} . In Fig. 1, area *A* is of comparatively high risk and area *B* relatively low. Accidents represented by points falling in area *A* would not be

acceptable under Farmer's criterion, and steps would be taken to reduce the release magnitude or to decrease its probability of occurrence. This would, in effect, move such an accident into area *B*.

We agree with the probabilistic philosophy behind the limit-line concept and feel that additional work is needed to define the location of the limit line more specifically. This is done when the concepts of individual mortality risk, site meteorology, and medical response to radiation are used to quantify the limit-line concept.

MORTALITY RISK ANALYSIS AND THE LIMIT-LINE CONCEPT

Individual Mortality Risk from a Nuclear Plant

Table 1 lists accidental-death statistics in the United States for the year 1967, the most recent statistics available.⁶ As indicated in the table, there is a finite amount of risk of death in almost any voluntary or involuntary activity. By studying the statistics of Table 1, Otway and Erdmann⁵ suggested a value of 10^{-7} per person per year as the upper limit for mortality risk incurred by a person living at the exclusion distance of a nuclear plant.* This figure is comparable with minor risks to which people are exposed and which they implicitly accept; for example, lightning. In this article we therefore assume a mortality risk of 10^{-7} per person per year as an acceptable value.

Mortality Risk from the Farmer Limit Line

Let us calculate the total individual mortality risk per year from all accidents falling on the limit line. Parameters taken into account include the amount of ^{131}I released, site meteorology, probability of wind damage, probability of induction of thyroid cancer as a result of inhalation of ^{131}I , probability of induction of leukemia from deposited ^{131}I on the ground, and probability of death from such diseases. The analysis will be made for two cases of wind condition: (1) an equal probability of wind blowing in any direction around the plant and (2) the more pessimistic assumption of a prevailing wind blowing in one 30° sector at the plant site.

Previous analysis by Beattie³ of the consequences of volatile fission-product releases assumes that control measures are taken to prevent the consumption of

*The same 10^{-7} value was suggested also by Willis.⁷

contaminated food (milk). Only the inhaled dose to the thyroid and the whole-body dose from ground-deposited iodine are considered in Ref. 3. Although the ingestion of radioactive material through food intake

Table 1 Accidental-Death Statistics for 1967

Type of accident	Total deaths	Probability of death per person per year
Motor vehicle (M.V.)	52,924	2.67×10^{-4}
Falls	20,120	1.0×10^{-4}
Fire and explosion	7,423	3.72×10^{-5}
Drowning	5,724	2.86×10^{-5}
Firearms	2,896	1.45×10^{-5}
Poisoning (solids and liquids)	2,506	1.25×10^{-5}
Machinery	2,055	1.0×10^{-5}
Aircraft	1,799	9.0×10^{-6}
Inhalation and ingestion of food	1,607	8.0×10^{-6}
Poisoning (gases and vapors)	1,574	7.8×10^{-6}
Water transport	1,545	7.7×10^{-6}
Blow from falling or projected object or missile	1,435	7.2×10^{-6}
Accident in therapeutic medical and surgical procedures	1,262	6.31×10^{-6}
Mechanical suffocation	1,182	5.9×10^{-6}
Foreign body entering orifice other than mouth	1,117	5.59×10^{-6}
Railway accident (except M.V.)	997	5.0×10^{-6}
Electric current	992	4.97×10^{-6}
Hot substance, corrosive liquid, steam	376	1.88×10^{-6}
Inhalation and ingestion of nonfood object	373	1.81×10^{-6}
Excessive cold	327	1.63×10^{-6}
Road transport accident (except M.V., rail, streetcar)	283	1.4×10^{-6}
Hunger, thirst, exposure	193	9.7×10^{-7}
Accident in nontherapeutic medical and surgical procedures	169	8.5×10^{-7}
Cataclysm	165	8.3×10^{-7}
Cutting and piercing instruments	149	7.47×10^{-7}
Animals (nonvenomous)	125	6.3×10^{-7}
Excessive heat and insolation	96	4.8×10^{-7}
Vehicle (nontransport)	96	4.8×10^{-7}
Lightning	88	4.4×10^{-7}
Bites and stings—venomous animals and insects	44	2.7×10^{-7}
Explosion of pressure vessel	42	2.1×10^{-7}
Lack of care of infants	22	1.1×10^{-7}
High and low air pressure	13	7.0×10^{-8}
Streetcar (except M.V. and train collision)	5	2.5×10^{-8}
Radiation	0	0
Late effect of accident (death more than 1 year after accident)	789	3.95×10^{-6}
Other and unspecified	2,656	1.33×10^{-5}

can be a very real danger, it can be greatly mitigated through the control and confiscation of contaminated foods. We have assumed such a control activity to have taken place in our analysis. It is, of course, not possible to control the atmospheric movement of radioactive material, nor the breathing rate or whole-body exposure to the environment. Hence we have considered these risks to be the more serious when compared to that of ingestion.

A single nuclear accident is represented by a point on the Farmer¹ limit line *b* of Fig. 1. The mortality risk as a consequence of this release for a person living at the exclusion radius of the plant is then given by

Individual mortality risk = received radioactive dose (in rems) $\times$ probability of induction of thyroid cancer per rem of dose $\times$ probability of death once thyroid cancer is contracted $\times$ probability of wind damage + received radioactive dose (in rems) $\times$ probability of induction of leukemia per rem of dose $\times$ probability of death once leukemia is contracted $\times$ probability of wind damage. (1)

The probability of thyroid-cancer induction by ^{131}I is reported by Dolphun and Marley⁸ as being between 10 and 30 cases per 10^6 man-rems. We shall use the average figure of 20 cases per 10^6 man-rems. We also assume (first) an equal probability for the wind direction from the plant. Thus the probability of wind blowing in any 30° sector around the plant will be equal to $\frac{30}{360} = \frac{1}{12}$.

In 1961 Pasquill⁹ proposed a method to estimate plume diffusion when detailed wind data were not available. He expressed the height and the angular spread of a diffusion plume in terms of more commonly observed weather parameters. He determined curves of height and angular spread as a function of distance downwind for several tropospheric-stability configurations. In 1961 Gifford¹² converted Pasquill's values of angular spread and height into standard deviations of plume-concentration distribution, σ_y and σ_z . However, Turner¹³, in 1969, used Pasquill's method for atmospheric-diffusion estimates since he had concluded that this work formed the basis of the standard-deviation approach as well. We have chosen to follow Turner in this regard. Alternate meteorological methods could be incorporated into our analysis; they would not detract from the utility of our siting approach.

To calculate the ^{131}I dose (in rems) from a release of R curies at a distance d downwind from the plant, use the following Pasquill dispersion formula:⁹

$$C_0 = \frac{168}{u d \theta h}$$

where u = wind speed, m/sec

h = plume height, m

θ = lateral cloud spread, deg

d = distance downwind, m

C_0 = dose-rate density, Ci/m³

We have made the conservative assumption that no decay occurs during passage of the cloud.

This dose-rate density is from a release rate of 1 Ci/sec. For a total release of R curies, the value of the total dose density will be scaled accordingly. Assuming that inversion weather conditions (category F) exist and that θ varies linearly from 30° at 100 m to 15° at 10,000 m, we can calculate the centerline dose density [Ci/(sec)(m³)] as

$$C = 9.36 \times 10^{-6} R \quad (2)$$

For a source strength of R curies, where we have assumed $d = 300$ m (assumed exclusion radius for the plant), $u = 2.0$ m/sec (for the class F stability category) $h = 12$ m (from Pasquill,⁹ Fig. 2), and $\theta = 29.7^\circ$. The exclusion radius selected relates the limit line to the individual mortality risk. We chose a 300-m radius as the nominal size for a reactor site in this analysis. Table 2 was utilized (from Beattie⁴) to transform this concentration of radioactive iodine into induced individual dose in rems.

Since any population consists of all age groups, a population-weighted concentration is used to give a 1-rem thyroid dose of 1.326×10^{-3} Ci/(sec)(m³) to an "average person." When Eq. 2 is combined with this value for the concentration-dose relation, the thyroid dose induced in an average person from a release of

R curies of ^{131}I under the specified weather conditions is derived as

$$\frac{9.36 \times 10^{-6} R}{1.326 \times 10^{-3}} = 0.705 R \quad \text{rems}$$

Reference 5 gave the mortality probability from thyroid cancer as 10% for children and 50% for adults. The weighted average is thus 44% using the population distribution shown in Table 2.

Now, utilizing Eq. 1, we can show that the individual mortality risk from thyroid cancer is

$$0.75 R \times \frac{20}{10^6} \times \frac{44}{100} \times \frac{1}{12} = 0.554 \times 10^{-6} R$$

The procedure in Refs. 3 and 10 is used to calculate the whole-body dose from ground-deposited ^{131}I . From Table 8 in Ref. 10, the time integral of gamma activity of deposited ^{131}I is 9.7×10^{10} Ci/(MeV)(sec) per thermal megawatt. Therefore the time integral of gamma activity of deposited ^{131}I is

$$\frac{9.7 \times 10^{10}}{25,000} = 3.88 \times 10^6 \text{ Ci/(MeV)(sec)} \\ \text{per curie of } ^{131}\text{I}$$

since ^{131}I has a half-life of 8 days and an equilibrium concentration of 25,000 Ci/MW(t).

Utilizing Table 7 of Ref. 10, we find that the dose due to surface contamination is 4.5×10^{-3} rad/(Ci)(MeV)(sec) per square meter. From Ref. 3, Fig. 4, for a deposition velocity of 3.0×10^{-3} m/sec at a distance of 300 m for class F weather, the ground deposition is 3×10^{-6} Ci/m² per curie released. Thus for a release of R curies of ^{131}I , the time integral of gamma activity of ground deposition is $3.88 \times 10^6 \times 3 \times 10^{-6} R = 11.4 R$, Ci/(MeV)(sec) per square meter. Hence the time integral of gamma dose 1 m above the ground is $11.4 \times R \times 4.5 \times 10^{-3} = 5.12 \times 10^{-2} R$, rads, = $5.12 \times 10^{-2} R$, rems, since the RBE factor for gamma rays = 1.0.

The probability of induction of leukemia⁵ is 30 cases per 10^6 man-rems, with a mortality probability of 100% for all age groups. Then individual mortality risk from leukemia is

$$0.0512 R \times \frac{30}{10^6} \times \frac{100}{100} \times \frac{1}{12} = 0.128 \times 10^{-6} R$$

Thus the net individual mortality risk from the reference accident becomes $0.554 \times 10^{-6} R + 0.128 \times 10^{-6} R = 0.682 \times 10^{-6} R$.

Table 2 Concentration-Dose Relation

Age range, years	Fraction of population, %	Concentration of ^{131}I needed to give 1 rem to thyroid, 10^{-3} Ci/(sec)(m ³)
0 to 5	8	0.52
5 to 10	8	0.76
10 to 15	8	1.04
15 to 20	8	1.36
20 to ∞	68	1.52

To arrive at *total* individual mortality risk per year from the spectrum of accidents represented by the limit line, we must integrate over all accident probabilities, as shown in Fig 2. Thus total individual mortality risk per year is

$$\int_{y=10^{-9}}^{y=3(10^{-2})} 0.682 \times 10^{-6} R(y) dy \quad (3)$$

Fig. 2 Approximated Farmer limit line.

The change in slope of the Farmer limit line between the 10^{-3} and 10^{-2} probabilities was approximated by a straight line, as shown in Fig. 2. Since the equation for this limit curve is

$$R(y) = \begin{cases} 10y^{-3/4}, & 10^{-9} < y < 10^{-3} \\ 0.087y^{-1.36}, & 10^{-3} < y < 3 \times 10^{-2} \end{cases}$$

where y and R are in years^{-1} and curies, respectively, the integration yields the total individual mortality risk per year as 3.40×10^{-6} . Extending the lower limit in Eq. 3 to $y = 0$ does not materially change the result.

This calculated figure is 34 times higher than the individual mortality risk of 10^{-7} suggested earlier. We tried several new curves in the process of looking for a Farmer limit line, so that the resulting total risk would be about 10^{-7} . The curve shown in Fig. 3 is one such result. The equation of this curve is

$$R = 10^{-5/4} y^{-3/4}$$

and the integrated risk from this limit line was 1.1×10^{-7} . This curve then satisfies the Farmer limit-line probabilistic release philosophy and also the 10^{-7} value for individual mortality risk per year suggested in Ref. 5.

Fig. 3 Acceptable limit line in the case of equally likely wind direction.

A similar analysis was repeated for the case of a site having a prevailing wind condition over a 30° sector, i.e., a probability of wind damage of 1.0. Under such an assumption, possibly pessimistic, the total individual mortality risk per year as calculated from the original Farmer curve *b* of Fig. 1, is

Total individual mortality risk per year

$$\begin{aligned} &= \int_{10^{-9}}^{10^{-3}} 8.2 \times 10^{-6} \times (10y^{-3/4}) dy \\ &\quad + \int_{10^{-3}}^{10^{-2}} 8.2 \times 10^{-6} \times (0.087 \times y^{-1.36}) dy \\ &= 4.1 (10^{-5}) \end{aligned}$$

This risk figure is 410 times higher than that of the target 10^{-7} value.

To approach our desired 10^{-7} risk objective for this case of a prevailing wind condition, relocation of the original Farmer limit line resulted in the curve shown in Fig. 4. The equation of this curve is given by

$$R = 10^{-1.24} y^{-3/4}$$

and we find that

Total individual mortality risk per year

$$\begin{aligned} &= \int_{10^{-9}}^{4.2 \times 10^{-4}} 8.2 \times (10^{-1.24} y^{-3/4}) dy \\ &= 1.015 \times 10^{-7} \end{aligned}$$

Fig. 4 Acceptable limit line in the case of a prevailing wind.

CALCULATED NUCLEAR PLANT ACCIDENT DATA COMPARED WITH THE LIMIT LINES

A probabilistic safety analysis of three nuclear plants has been carried out and published by different researchers. The three plants are (1) the Windscale Advanced Gas-Cooled Reactor (AGR),¹ a CO₂-cooled

graphite-moderated reactor; (2) a nominal 1000-MW(e) pressurized-light-water reactor⁵ scaled from the Crystal River Plant; and (3) the Swiss NOK Benzau nuclear plant,¹¹ a pressurized-light-water reactor.

Results of these investigations are shown in Fig. 5. Curve 1 represents the Crystal River Plant, and sets of data are also shown for the AGR and NOK Benzau plants. The original Farmer limit line is plotted as curve 2, and the second modified limit line determined by us is shown as curve 3. The accident analysis for the Benzau plant assumes that a release of radioactivity occurs only when the containment is breached. In actuality a release can occur without a vessel breach; hence the data are not particularly useful.

To offset statistical uncertainties in the failure-probability data, the release-probability curves for a nuclear plant could be shifted to allow for data uncertainty. Such an approach was outlined in Ref. 5. In any event, one observes that most of the contribution to the total risk comes from the low-release high-probability accidents.

One method of analyzing the accidents of nuclear plants, suggested by Otway and Erdmann,⁵ is to sum up the risk from all accidents and compare it with the

Fig. 5 Limit-line comparison with estimated plant accident data.

10^{-7} value. A more stringent criterion would be not to permit any of the accident points to fall above the limit line since the accident points are discrete and nothing is known about the probability of unknown accidents between the known discrete ones. We can see that, if any of these plants had a 300-m exclusion radius, those accidents represented by points falling above the new limit line would not be permitted.

Of course the above conclusions are based upon a simplified analysis of a reactor and a site. In order to carry out an actual limit-line safety analysis, one would have to analyze the site very carefully from the meteorological and geological points of view, as well as with respect to the reactor and its associated safeguards. We did not do this in the above analysis of course, as we were attempting to present a formulation. In addition to site meteorology and geology, the actual population distribution becomes very important in determining probable risk. Some dissertation research has been carried out on more realistic site-reactor combinations at UCLA.

CONCLUSIONS

We calculated the individual mortality risk per year for persons living at the 300-m exclusion radius of a nuclear plant using the original Farmer limit line. For an equal probability of wind direction, we derived a value 34 times higher than the target, 10^{-7} ; in the case of a prevailing wind direction, it was 410 times higher. For each wind condition a modified limit line was constructed to meet the 10^{-7} value. Should the exclusion radius be varied, a different limit line would have to be constructed.

This analysis was carried out to determine the risk to a person located at the site boundary because of an improbable spectrum of accidents lying along a limit line. A more appropriate analysis would be one that included local demography and unique site characteristics. Such factors as prevailing winds and types of released fission products could then be included properly, without implying questionable conservatism in final results obtained by incorporating these factors

in an artificial manner. In this work we assumed no fission-product decay during the inadvertent release situation, and we have completely neglected emissions during normal operations.

Our short sampling of the nuclear plant probability analyses done to date indicates that a limit-line safety approach is feasible with presently designed plants. Moreover we can relate, very simply, the limit-line data to an integrated risk value by factoring in weather data and exclusion radii. Certainly an approach based on the modified limit-line concept provides us with a great deal of insight into power-plant behavior that can help us design our plants more safely.

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4. NUCLEAR SHIP CRITERIA

4.1 Nuclear Merchant Ship Site Criteria

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[*Nucl. Safety*, 4(2): 27-33 (December 1962)]

The safeguards review of the N.S. *Savannah* by the Advisory Committee on Reactor Safeguards (ACRS) and the Division of Licensing and Regulation was concluded during the summer of 1962 with authorization, on August 3, for interim operation extending over a six-month period and covering approximately six port areas. As a consequence of the study which led to this authorization, the Maritime Administration-AEC Joint Group developed a guide for the "Port Operation of Shipboard Reactors."³⁰ The principal part of this guide modified the AEC Reactor Site Criteria³¹ in such a manner as to make some allowance for the mobility of the ship reactor and to arrive at criteria which would be compatible with both the inherent aspects of nuclear merchant vessels and the safety philosophy that is expressed in the existing site criteria.

Essential Features of Ship Site Criteria

One of the most important developments was the adaptation of a criterion for total population exposure in lieu of the city distance stipulation in the existing guide.³¹ As an interim measure, a criterion of a permissible exposure dose of 2,000,000 man-rem was established, as had previously been proposed.³² This exposure is in addition to the individual exposure limit of 25-rem whole body (and 300-rem thyroid) that is employed for calculating various defined zones around the ship berth site in a manner similar to the zones around a stationary reactor site.

Other unique features of the ship site criteria involve the removal of the vessel after an acci-

dent (as a means of limiting the total population exposure), the manipulation of the ship reactor power history to match the capability of the berth, and provisions for the exposure potential during the time the ship is entering and leaving a port.

The guide³⁰ developed for mobile reactors, such as the N.S. *Savannah*, established three zones that vary in distance from the reactor, as follows:

1. Controlled Zone—that area in which all persons are under the direct control of ship's personnel so that in the event of an accident evacuation could be effected within two hours without any person receiving more than 25 rem whole body or 300 rem thyroid, and any individual general public member standing at its outer boundary for two hours would not receive an exposure exceeding 25 rem whole body or 300 rem thyroid.

2. Low Population Zone—that area in which it is reasonable to expect that, in the event of an accident, total evacuation or protective measures could be carried out in a graded fashion within 24 hours so that no person would receive more than 25 rem whole body or 300 rem thyroid, and any member of the general public standing at its outer boundary for 24 hours would not receive an exposure exceeding 25 rem whole body or 300 rem thyroid.

3. Dense Population Zone—that area which is immediately adjacent to the outer boundary of the low population zone and cannot be evacuated, controlled, or protected.

It can be seen that the establishment of these zones is not only a function of the fixed port environment, but also a function of the potential reactor accident. If the port environment remains fixed,

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the reactor's potential maximum accident can be adjusted. Therefore, the basic requirement for suitability for any berth is that the reactor accident potential be adjusted by controlled reactor power history so that the zone boundaries for exposure limitation are equal to or less than those boundaries established on population control or ship mobility.

To ensure appropriate limitations to long-term effects on the population as a whole, the criteria set forth a 2,000,000 man-rem whole body limitation on the exposure to the total population in all three zones surrounding the reactor for the duration of the accident. Because of the inherent mobility of a shipboard reactor, the duration of the accident is taken to be 24 hours maximum.

It is this very mobility, however, that makes it necessary to consider the varying reactor conditions and the varying position of the reactor with respect to the port environment. Thus, irrespective of an acceptable zone boundary based on individual exposures, the total population dose level should also be met at all times.

Other Aspects of the Criteria

In view of the many unique differences between siting a stationary power reactor and berthing a nuclear merchant vessel, the ship site criteria set forth in some detail the conditions for port entry and the arrangements that must be made. Such conditions and arrangements implement the general objectives of the criteria and ensure their attainment.

The specified conditions for port entry³⁰ include the following:

1. Invitation from port authorities
2. Evaluation of environmental conditions
3. Evaluation of port emergency capabilities
4. Notification of all organizations involved
5. Screening of suitable berths
6. Evaluation of access and egress routes
7. Preparation of detailed accident analysis
8. Availability of tugboats
9. Testing of emergency equipment
10. Approval of emergency plans
11. Limitations on shipboard visitors
12. Reporting and approval of above factors

In addition to the above conditions for port entry, the ship site criteria³⁰ give 24 "additional measures applied to initial operations." These additional measures are specific provisions with respect to the calculation of the specified zones or operating procedures and accident as-

sumptions that would, in effect, introduce extreme safety in the operation of the vessel and extreme conservatism in the calculation of the exposure that might result from the maximum credible accident. The additional measures to be applied for initial operations are listed below:

1. Use of actual reactor power history
2. No credit to be taken for emergency cooling
3. Assume 50% iodine plateout in the containment vessel only
4. Containment-vessel leakage rate assumed to be 2.5% per day
5. Iodine filter efficiency assumed to be 99%
6. No credit to be taken for variable wind direction over the 24-hr period
7. No credit to be taken for other than inversion conditions over the 24-hr period
8. Dispersion calculations based on 1 m, sec wind speed, $C_v = 0.4$, $C_v = 0.07$, $n = 0.5$, $h = 0$
9. Specific emergency plan to be established
10. Auxiliary boilers to be tested in port
11. Vacuum to be maintained on main condenser
12. Ship visitors to be limited to one-half the acceptable number
13. Site visitors to be limited to one-half the acceptable number
14. Evacuation drills to be conducted regularly on ship and on site
15. Reactor compartment integrity and emergency system operability to be tested periodically
16. Containment-vessel leakage rate, filter efficiency, and duct tightness to be tested periodically
17. Senior nuclear staff member to be aboard ship in port
18. Health-physics staff member to be aboard at all times reactor is operating
19. Port analysis report to be approved by regulatory authorities
20. Conservative zone boundaries for 25 rem in specified time: 1 hr, 480 ft; 2 hr, 650 ft; 24 hr, 1600 ft
21. Instantaneous release of fission products to containment vessel
22. Total population exposures assume maximum number of persons in controlled zone
23. Individual exposures less than 0.05 rem not considered

24. No credit taken for holdup in the reactor compartment

Discussion

The existing "Reactor Site Criteria" were specifically intended to reflect stationary-reactor siting practice in this country. The criteria not only did not cover many of the new problems introduced by a mobile reactor but also would have specifically excluded nuclear shipping from most harbors. As has been previously noted,^{33,34} the provision that made it impossible to apply the stationary-reactor criteria to mobile reactors was the stipulated city distance, i.e., "at least $1\frac{1}{2}$ times the distance from the reactor to the outer boundary of the low-population zone." Since the specified low-population-zone boundary increases from 1.6 miles for a 60-Mw(t) reactor to 4.5 miles for a 300-Mw(t) reactor, it is readily apparent that city distances $1\frac{1}{2}$ times the above would deny a nuclear merchant ship access to most ports in the world. This being the case, it is important to know what the function of the city-distance criterion was, as stated in the "Reactor Site Criteria":³¹

One basic objective of the criteria is to assure that the cumulative exposure dose to large numbers of people as a consequence of any nuclear accident should be low in comparison with what might be considered reasonable for total population dose. Further, since accidents of greater potential hazard than those commonly postulated as representing an upper limit are conceivable, although highly improbable, it was considered desirable to provide for protection against excessive exposure doses to people in large centers, where effective protective measures might not be feasible.

Although both these objectives are not readily achievable by a single criterion, the mobile-reactor site criteria affect the former by use of a total-population-exposure limit and the latter by safety features in the design, emergency procedures, and the special arrangements for port operation.

Once having established reasonable criteria, it is most important to demonstrate that the criteria can actually be attained in all circumstances. Of particular concern is the capability of removing the ship from its berth in less than 24 hr. Many studies have been made of the exposures as a function of time and posi-

tion following the maximum credible accident. These are formally summarized, so far as the N.S. Savannah project is concerned, in the Port Analysis Reports (for example, the reports for Savannah, Ga.,³⁵ Norfolk, Va.,³⁶ Panama Canal,³⁷ and in an ORNL report,³⁸ which presents in more detail the probable exposures, independent of any port or berth conditions). From these studies it is concluded that the exposures to essential shipboard and tugboat personnel would be less than 1 r/hr submersion and less than 80 r thyroid inhalation for the several hours that might be required for the emergency removal of the ship, even if all personnel remained downwind of the release during that time.

The ship criteria also have provided for the relatively large number of persons who might be on site (i.e., within the controlled zone) as well as for those who may be aboard ship. In both instances these persons must be evacuable before they have received 25-rem whole-body or equivalent exposure. Their evacuation must be possible in the specified time, i.e., less than 2 hr, or their presence will not be permitted. In this connection it is noted that, even assuming the pessimistic leakage and release rates indicated in the interim criteria, the close-in exposures are substantially less than in stationary power plants because of the double containment; thus the planned evacuation of personnel is considered feasible.

It appears that the mobile-reactor criteria have provided a reasonable extension of the Commission's "Reactor Site Criteria" to meet the needs of ship reactors. In so doing, it has introduced or at least extended certain concepts that will assume greater significance in siting considerations as nuclear technology develops. In addition to the total-population-exposure limit, the following concepts are implicit or explicit in the ship criteria:

1. Controlled release of activity from the accident
2. Dependence on operating procedures to minimize the consequences of the maximum credible accident
3. Dependence on administrative control to assure the integrity of certain features (i.e., reactor compartment)

The fact that each of the concepts is involved in conditions that were authorized for the N.S.

Savannah does not imply approval of the concepts themselves but rather that, given justifiable reasons, modifications to existing practice will be considered. This is really no more or less than the approved "Reactor Site Criteria"³¹ (for stationary reactors) pointed out in the statement that a "unique or universal feature having a significant bearing on the probability or consequences of accidental release of radioactive materials ... would be taken into consideration."

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4.2 Port Surveys for Nuclear Ships

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[*Nucl. Safety*, 5(4): 321-323 (Summer 1964)]

An elaborate set of guides^{44,45} has been developed by AEC to govern the placement of stationary nuclear power plants with regard to inhabited areas. These rules seek to safeguard the population at large by requiring a combination of engineered safety features and remoteness of location to minimize, respectively, the emission of radioactivity and the radiation exposure to nearby human life in the event of a reactor accident. Because of the permanently fixed location of such reactors, the initial haz-

ards analysis of the reactor site should remain valid for the life of the plant, barring sizable population or power-level changes.

Mobile-Reactor Site Criteria

The operating requirements of a nuclear merchant ship present a more complex problem in safeguards analysis. To perform its function, such a vessel must approach and temporarily

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remain within many different population centers at berths offering widely disparate degrees of remoteness from habitation. In compensation for this practical limitation to siting, however, the nuclear vessel has the ability to move or be moved to a remote site following a nuclear accident within a time sufficiently short to mitigate substantially, if not preclude, population exposure. It is on this basis, therefore, that the potential hazards associated with the visit of a nuclear ship are examined.

The site limitations for an acceptable N.S. *Savannah* berth have been derived from dose calculations by using the maximum credible accident concept in a manner comparable to that employed for land-based reactors. By taking into account the fission-product inventory at the time of arrival and the effectiveness of engineered safeguards, based on the routine testing used to verify their operability, one can estimate the potential activity release following a maximum credible accident.⁴⁶ With such estimates as a starting point, the dose receivable by individuals exposed for various lengths of time at different distances from the ship and under various meteorological conditions can be computed. Various constraints are then applied to the computed results to assure that individual doses would not only remain below the hypothetical limits but that all doses of whatever magnitude to the population at large would be minimized insofar as practical handling of the ship permits. The site limitations established by the dose calculations have been translated into a set of criteria^{47,48} acceptable to government regulatory authorities for (1) evaluating the suitability of the available berths in a port, (2) determining the time within which the ship must be removed to a remote anchorage following an accident, and (3) selecting a suitable remote anchorage.

Suitable Berth Criteria

In practice, other constraints affect the choice of berth in addition to the nuclear safety considerations cited above. For example, the berth must be of suitable size and offer adequate draft alongside to accommodate the vessel; its facilities must be capable of handling the flow of cargo, passengers, and visitors; its location must permit ready and economical approach and egress for goods and personnel, while at the

same time providing easy access for emergency vessels and minimal interference with port traffic if an accident should require sudden removal of the ship; and the site must be unoccupied and available for the period of the projected visit.

Fortunately, several of these considerations are qualitative in nature and may be open to compromise, depending on the type and quantity of the cargo, the number of passengers aboard, and concessions by local traffic-control authorities. Berth size and nuclear safety aspects involve definite minimum requirements and hence receive primary consideration.

The population-distance criteria also actually permit a certain flexibility of choice. Within limits, proximity of population to the dock site may be compensated for by reducing the fission-product burden of the reactor; this is accomplished by limiting the reactor power over a period prior to port entry. In all cases, however, tugs must be held alongside the ship as an additional guarantee of prompt removal following an alert, until radioactive decay lowers the fission-product inventory to a level which would delay fuel melting for 1 hr following loss of cooling.⁴⁵ The purpose of the port survey is to select and to arrange for the use of the most suitable berth available, subject to the above limitations.

Survey Team

The N.S. *Savannah* port surveys⁵⁰ have been performed by a team comprising a ship's captain and a nuclear engineer. The pair commonly initiates a port survey by arranging a joint meeting with administrative and regulatory officials from appropriate local, state, and federal bodies. The officials are briefed concerning the berth requirements and emergency procedures; arrangements are made to examine the docks; data for ship's communication are gathered; and an emergency alert list is compiled. After the docks and their environs are inspected and a suitable berth is secured and locally approved, operational details such as time of arrival, escort vessels, tugboat service, pilotage, and location of remote anchorage are resolved; all affected parties are advised; and a survey report is prepared and submitted to the AEC-Maritime Joint Group. This report, with suitable background material appended, is

forwarded to the AEC regulatory staff for approval prior to the ship's arrival. A number of such reports have been prepared and have provided the basis for subsequent entry of the N.S. *Savannah* into 10 U. S. ports.⁵¹⁻⁵²

Foreign Ports

In the Spring of 1963, the first surveys of foreign ports for N.S. *Savannah* visits were conducted in several northern European countries. All the nations involved had expressed prior interest in receiving the ship, and formal agreements with the United States concerning the visits were either completed or approaching conclusion at the time of the surveys. While the AEC criteria were employed as a basis for the surveys, local nuclear shipping regulations were also in effect in some ports. Since these had not previously been exercised, considerable discussion was conducted to formulate specific interpretations and clarify the intent of the wording as applicable to the *Savannah*. Varying nuclear safety aspects are stressed in different ports, reflecting the diverse effects of national area, population size, economic bases, and geographical situations. A particular effort was made to avoid subjecting the ship to measures restricting its freedom of navigation, on the basis that it should be entitled to the same treatment as any other ship of its tonnage and seaworthiness ratings while under way in open waters.

Port Criteria in Practice

In spite of the often divergent requirements for port entry of a nuclear merchant ship, a satisfactory resolution can usually be achieved if sufficient incentive exists. This was adequately borne out by the series of 11 domestic port calls⁵¹ made by the *Savannah* in 1962 to 1963. Several of the berth accommodation considerations involving economic aspects (e.g., maintaining extra tugboats on extended standby, rearranging berthing assignments, or reserving space that might otherwise have gone to more commercially productive allocation) were fulfilled through extra expenditures or through the extraordinary cooperation of private and public officials in the interest of bringing the N.S. *Savannah* into their ports on her maiden tour. At such future time as nuclear shipping becomes common, berthing restrictions could create an

economic disadvantage, in comparison with the fossil-fueled vessels, particularly for general cargo and passenger ships, unless these restrictions can be moderated through judicious use of engineered safeguards and a more sophisticated appraisal of their effectiveness by regulatory organizations. There is increasing evidence, for example, that on the order of 90% of the iodine released in a reactor accident would plate out on the containment system walls and ducts before reaching the atmosphere.⁴²⁻⁴⁴ With further work in this field, it may become reasonable to take credit in dose calculations for this attenuated release. Similarly, the short-term dose-ameliorating effects of multiple-containment holdup⁴⁵ will undoubtedly be exploited by reactor designers as soon as the worth of this technique is acknowledged in the rules of licensing organizations.

In time, successful performance records and increased confidence in nuclear plant reliability should permit the establishment of simple uniform rules for nuclear ship entry. For example, docks might be periodically certified by an appropriate regulatory body as satisfactory berths for nuclear ships of specified ratings, and standard emergency procedures might be developed. These provisions would obviate the need for the port-by-port, visit-by-visit negotiations now necessary. During the present infancy of nuclear shipping, the port survey provides a means, however cumbersome, to permit the accumulation of operating experience that can ultimately justify the removal of penalizing restrictions.

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5. NOVEL SITING CONCEPTS

5.1 Underground Containment of Power Reactors

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[*Nucl. Safety*, 2(3): 31-34 (March 1961)]

A nuclear power reactor operating for an extended period of time will build up a sizable fission-product inventory that could cause disastrous loss of life and property if released. For this reason, most power reactors are provided with a containment envelope independent of the rest of the power system so that failure of the system will not result in excessive activity release. The normal procedure¹ is to provide a vapor-containment pressure vessel, designed to low-leakage specifications, along with an exclusion area surrounding the plant, which prevents inhabitants of areas near the plant from being exposed to excessive radiation in case of an accident. The exclusion area is needed to reduce the direct gamma irradiation at the boundary site and to allow sufficient air dilution of fission products leaking from the containment vessel.

In some countries the purchase of exclusion areas is extremely expensive, and in some cases it is difficult to find acceptable sites that are sufficiently distant from large inhabited areas. This is partly because thermal power plants require a heat sink, which usually stipulates a site near a river, and the bulk of the population is usually settled along rivers. In Switzerland, for instance, if a 10-km circle is drawn around each settlement with a population greater than 10,000 inhabitants, it is found that almost all the river sites are overlapped.² Thus any scheme that eliminates the necessity of an exclusion area would be advantageous in this case. Furthermore, land acquisition prices would be reduced, along with power transmission costs. Underground construction of nuclear power plants could provide a solution to this problem, even though the actual cost of such construction would usually be higher.

Although in more sparsely settled countries it would be more expensive to build a plant underground, almost complete invulnerability to military attack could be attained. This facet of the problem is usually overlooked when decisions are made on a strictly economic basis, but the fact remains that the supply of power is one of the prime prerequisites to reconstruction after military destruction.

Underground Construction

Underground construction, as discussed here, specifically refers to installation of the power plant in a cave or in excavated rock. There has been a great deal of experience on tunneling and other facets of underground construction. Norway has actually built the Halden nuclear plant in a rock excavation,³ and many hydroelectric power plants have been built underground on a strictly economic basis. From a technical point of view, there are several advantages of a rock-contained plant. Among the advantages is the fact that the internal pressure in the chamber caused by an accident would be taken up wholly or, at least, in part by the rock. The reactor chamber could be readily located so that no fragments of an exploded plant could reach the surface of the ground and thus create a direct path for the release of fission products to the atmosphere. The possibility of destruction of the plant from without would be decreased. Fission products in the form of gases or aerosols would have to seep through rock before being able to reach the surface of the ground, and in the process they would lose activity since (1) there would be natural decay during the time required to seep through the

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rock, (2) some fission products would be retained in the rock and in the overburden soil by ion exchange, (3) aerosols and vapors would be arrested on damp surfaces, and (4) inert gases, such as xenon or krypton, would decay after a relatively short period of time into products that could be absorbed by the rock and the soil overburden.

However, there are disadvantages. The costs of underground installations, although they may be competitive in foreign countries, have been estimated to be from 3 to 7 per cent of the total plant cost greater than the costs of an equivalent surface installation.⁴ Further, it is not always possible to ensure ahead of time that the rock quality will be good throughout the volume of excavation and that heavy flow of ground water does not exist; also it is not always possible to find an acceptable geological structure adjacent to a large body of water. Cooling towers could be used, but these are expensive and would have to be placed outside the protective area. Ground water may be contaminated in case of an accident concurrent with rock shifting or a slight movement which might cause a fissure of the excavation lining. Finally, there would be limited flexibility with respect to access and expansion. A nontechnical advantage of the underground installation is that the surface topography would be unchanged and would thus maintain its natural beauty.

Some unique requirements must be met for a site to be acceptable for underground construction. The rock formation should be easy to excavate but should be structurally sound. It would be advantageous if the topography were such that a horizontal entry could be made, i.e., from a side of a hill. Existing openings, such as mines and caves, could be used at a great saving in cost if they were adaptable to the requirements. Such sites have been surveyed for underground installations.⁴

It appears that limestone, granite, sandstone, and salt formations have the best properties for excavation.⁴ These formations usually permit wide spans, high ceilings, and regular pillar arrangement. Limestone is the primary geological formation used in the United States for underground installations. The limestone deposits in the United States are generally horizontally bedded, and many sites are desirable because they permit horizontal entry, large flat roof spans, and high ceilings. Granite gneiss

deposits are second in importance. Most of the investigated granite formations are accessible by horizontal entry from the sides of hills or cliffs. Granite formations usually permit large arched roof spans and high ceilings. Sandstone formations are in many cases adaptable for underground construction, but possible sites are not so abundant as those in limestone or granite. The average strength is less for sandstone than for limestone or granite, and its porosity is much higher. Salt deposits are favorable for underground construction and have the added advantage of dryness. However, entry must be by means of deep shafts. Where cost is not important, a plant could be installed in virtually any type of geological formation.

Excavations are usually made by drilling and blasting. The procedure is to drill a vertical entry to the top of the planned chamber. The roof section is excavated first, and "dental work" repairs are made to the roof to assure structural integrity. This work consists of tying poor rock together by steel straps, bolts, and concrete and, if called for, lining the excavation with concrete. Once this has been accomplished, excavation can proceed downward to the required depth without the hazard of rockfalls from the ceiling.

There is a great deal of experience in underground construction of this type. As of 1957, over 240 hydroelectric powerhouses had been built underground. Of these, the greatest number are in mountainous countries with reasonably advanced technology and relatively low labor rates; for instance, 44 are in Sweden, 53 in Norway, 36 in Italy, and 28 in Switzerland.⁵ The United States, however, has only four plants of this type but has had much experience in tunnels, mining, and other forms of excavation.

Containment in Underground Plants

The containment structure enveloping a nuclear reactor must withstand various loads are not dependent on whether the plant is aboveground or underground. For example, the internal pressure that would result from a loss-of-coolant accident or accidental chemical reactions in a pressurized-water or gas-cooled reactor plant would have to be successfully resisted by the structure. The escape of fission products would have to be prevented, as well as direct radiation damage to the nearby population, and measures would have to be taken to prevent breach of the enclosure by missiles.

The underground plant can be constructed to satisfy each of these requirements. Leakage of radioactivity could be positively prevented by a steel-sheet lining, or it could be reduced by various schemes. The chamber could be lined with concrete sprayed directly against the rock after excavation, in which case permeability of the concrete and overlying rock and the ion-exchange properties of the rock and overburden would be the determining factors in the release of gaseous contamination. Ground-water contamination could be controlled by placing chain tile around the reactor excavation leading to holdup tanks.⁶ This would reduce the water table in the vicinity of the reactor and provide for control of the release of radioactivity. A steel or plastic liner could be embedded in the concrete lining the rock walls, but unless the inner surface of the concrete was further lined with a nonabsorbing or removable substance, decontamination in the event of a minor accident would be difficult. The decontamination of bare rock would, of course, also be difficult.

An interesting scheme² is to line the rock walls with a course of load-bearing hollow blocks and to line the blocks with concrete. The annular space thus provided could then be pressurized so that leakage would be maintained into the enclosure from which it could be sent through filters and then discharged. The problems inherent in this system are similar to those of conventional building containment.⁷

If reliance is placed on the holdup time of the fission gases in the rock and ion exchange in the rocks and the soil, good knowledge of the underground water flow is required to avoid contamination of human water supplies. Absorption of fission products in various clays and rocks has been reported,² and the general problem was discussed in an earlier issue of *Nuclear Safety*.⁸ The relative effectiveness of the various materials in ion-exchange absorption of Sr^{90} is indicated in Fig. IV-1 (taken* from reference 2), in which the quantity of absorbed strontium in milligrams per 100 g of rock or clay is plotted against the initial concentration of strontium in the solution. These data were obtained by adding finely ground rock or clay to the solution and allowing it to react for $2\frac{1}{2}$ hr. The weight of the rock in the test

Fig. IV-1 Absorption of strontium in different clays and rocks.²

solutions was in the ratio 1:25. The strontium in solution was in the form of strontium nitrate. Figure IV-1 shows that the relative effectiveness in ion exchange of strontium is, for low concentrations, in decreasing order: bentonite, ordinary clay, Swiss bonding clay, French fire clay, kaolin, and sandstone. At higher concentrations, the order becomes: bentonite, sandstone, ordinary clay, French fire clay, Swiss bonding clay, and kaolin. With low strontium concentrations, complete absorption takes place with bentonite, whereas limestone absorbs less. Figure IV-2 (also taken from reference 2) shows similar data for the absorption of iodine in different clays and rocks. These data are for iodine in atomic form, dissolved in water with the addition of a small quantity of potassium iodide. Bentonite again shows the greatest absorption capacity. The data of both Figs. IV-1 and IV-2 indicate the hydrologic problems that exist in the field.

It is probable that a large absorption factor can be attained in natural rock if the crevices are not too large and, in addition, are moist.

* Figures IV-1 and IV-2 are reprinted² by permission from Comitato Nazionale per le Ricerche Nucleari, Rome.

Fig. IV-2 Absorption of iodine in different clays and rocks.²

The structure can be kept moist by proper growth of vegetation on the overburden soil.⁶

Cost of Underground Containment

The cost of containment constitutes a considerable portion of the total cost of a reactor installation. A hypothetical underground design for containment of the EBWR has been reported by Beck,⁴ in which the layout is in the same form as for the aboveground installation. In sound rock the 80-ft-diameter dome-shaped scheme should be feasible. For large openings in excess of 50,000 cu yd, Beck estimates the excavation cost to be possibly as low as \$8 to \$10 per cubic yard. For the reference EBWR design, it was estimated that the cost of the underground container would be approximately \$940,000; this cost includes rock excavation, concrete linings, tunnels, ventilation, and blast-proofing features. The over-all cost of the underground plant was found to be 7 per cent greater than the cost of the aboveground plant, as built.

The actual cost of the Halder underground reactor building, including the pretunnel, in Norwegian kronen, was as follows:³

Excavation of a volume of 8900 m ³ , including securing of rock surfaces before concreting	650,000
Concreting, including concrete shielding, air-lock walls, and painting	850,000
Total	1,500,000

The total civil-engineering cost amounted to 2,760,000 kronen. Aamodt³ estimates that an equivalent steel-shell structure in Norway would cost 2,950,000 kronen. At the official exchange rate of \$0.14 per kronen, this is about \$386,400 for the underground construction of an 11,650-cu yd volume. The construction costs come to about \$332 per cubic yard for the total work, including \$7.85 per cubic yard for the excavation. Of course, it is difficult to relate this to actual costs in the United States because of different labor rates and amounts of mechanization. Aamodt³ states that this is a relatively high cost because of the difficult transport of the excavated material from the lower parts of the plant, together with the rather extreme bolting of the cracked zones performed before the concreting. Beck,⁴ however, considers a cost of \$8 to \$10 per cubic yard quite a low figure, attainable only with large excavations; the actual cost in the United States at the present time probably varies between \$10 and \$30 per cubic yard.

Conclusions

It is clear that underground construction of reactors is feasible and, in some areas, may be competitive with aboveground construction of a steel containment shell. As an exclusion area becomes increasingly valuable, underground construction becomes more attractive. At the present time, it could be justified in some cases on the basis of invulnerability to military attack. Since, as mentioned above, a supply of power is one of the first requirements for reconstruction after a disaster, the 3 to 7 per cent increase in cost over that of the aboveground plant could be considered as insurance against total disaster.

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5.2 Offshore Siting of Nuclear Energy Stations

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Abstract: Questions involving environmental quality, priority of land uses and public safety have created and will continue to create difficulties in finding acceptable sites for nuclear power plants. Interest is increasing in offshore stations as one solution to the siting problem. Investigations of offshore stations have included both artificial islands and floating platforms and have touched on environmental effects, economics, and safety. Indications are that many of the problems associated with siting nuclear power plants could be alleviated by offshore stations. The floating-station concept, in particular, seems to have significant potential for the future. Before this potential can be realized, however, considerable development of the concept will be needed.

In recent years there has developed increasing public awareness and concern about the environmental consequences of a technological society. Serious environmental threats, as well as changing social values, have resulted in tighter government regulations and greater public participation in decisions concerning environmental pollution, public safety, and siting of industrial plants. These conditions, combined with the premium

value and general scarcity of suitable land, have resulted in near-crises in finding acceptable sites for electric power plants in some regions of the country. But the present problems are minor compared with those of the foreseeable future. The proportions of the future siting problem are indicated by the projected requirements for electric power in the United States. In the next 30 years, these requirements will increase from the present 300,000-MW(e) capacity to about a 2,000,000-MW(e) capacity. It is estimated¹ by 1990, 255 new sites for stations larger than 500-MW(e) capacity will be needed. Some generating stations to be built in the next 30 years will approach 10,000-MW(e) capacity.

The 30-year outlook for a sevenfold increase in total generating capacity and a large increase in station size has stimulated interest in nonconventional siting concepts. One such concept is the offshore station. The Japanese Ministry of International Trade and Industry is sponsoring investigations that are expected to lead to

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three 1000-MW(e) offshore nuclear stations.² Two of these stations are to be floating, and one is to be undersea. In the United States the Consolidated Edison Company of New York is examining offshore artificial islands for siting nuclear power plants.³ In addition to these recent investigations, for which little detailed information is available, there have been a number of other studies and some related experience. The purpose of this article is briefly to review these studies and to present some interpretive comments on the outlook for offshore stations.

SOURCE OF SITING PROBLEMS

Most of the difficulties in nuclear power-plant siting appear to arise from one or more of the following factors, and any proposal for nuclear power-plant siting must be examined in the context of these factors.

1. **Cost and Availability of Land.** With increasing population and per capita consumption of all industrial products, suitable land for industrial sites, including nuclear sites, is dwindling, and the price is increasing.

2. **Environmental Effects of Plant Effluents.** The consequences of normal discharges, particularly heat and low-level radioactive wastes, are often questioned. Waste heat can change the ecology of streams or alter the local "weather," depending on whether the immediate receptor is air or water. The validity of maximum permissible concentration values and the possibility of biological concentration are questions that arise with respect to radionuclides.

3. **Probability and Consequences of Accidents.** Accidents, whether they arise from plant malfunctions or from natural disasters, are of great concern. Of the natural disasters, earthquakes and the effects thereof on the nuclear plant are probably the most worrisome. In the event of an accident, the effectiveness of engineered safety features and the adequacy of the separation distance between the population and the plant are points that are questioned.

4. **Visual Presence and Other Sensory Effects.** The intrusion on the landscape of large shining domes, rectilinear sheet-metal buildings, cyclone fences, brick stacks, and other appurtenances of power stations are quite objectionable to many people, especially when these structures appear in a residential or naturally scenic setting. Other sights, sounds, and movements associated with a commercial operation are also sources of complaint.

EXPERIENCE WITH SHIPBOARD REACTORS

Experience with small pressurized-water reactors (PWRs) on ships, including the N.S. *Savannah*,^{4,5} the *Otto Hahn*,⁶ and numerous naval vessels, has been good. The U. S. Army Corps of Engineers constructed and has in operation a small 10-MW(e) barge-mounted nuclear power station.⁷ This PWR, designated MH1A, is now operating in the Panama Canal Zone. Such experience indicates that there are no inherent reasons why reactors could not be installed on floating platforms—at least not for PWRs.

STUDIES OF OFFSHORE STATIONS

With respect to large offshore nuclear power stations, there is no actual experience, but there have been a number of studies. These include

1. Investigations related to the proposed Bolsa Island nuclear power and desalting station, which was to have been located on an artificial island built $\frac{1}{2}$ mile off the coast of California near Los Angeles.⁸

2. An investigation by Arnold, Gall, and Morris of the feasibility of offshore nuclear dual-purpose plants.⁹

3. A study by Daniel, Mann, Johnson, and Mendenhall of an earthquake-resistant floating nuclear power station.¹⁰

4. An investigation by Klepper and Bell of underwater containment of nuclear power reactors.^{11,12}

5. A study by Bowers and Wichner of the economics of offshore floating nuclear power stations in comparison with inland stations using cooling towers.¹³

Bolsa Island Project

The Bolsa Island station was to have been a combination nuclear power and desalting complex with two 3000-MW(t) light-water reactors to produce 1800 MW of electricity and 150 Mgd of desalted water. Bechtel examined 17 coastal sites extending from Los Angeles to near the Mexican border, and at least three were judged to be satisfactory from the standpoint of safety and other siting considerations.⁸ However, comparative cost estimates based on appraised land values, cost of plant construction, cost of circulating-water marine lines, and the cost of desalted-water conveyance and electric transmission facilities indicated that a man-made island (Bolsa Island) was the most economical choice. The proposed island had

dimensions of 1500 by 1250 ft. or an area of about 43 acres. At the island site the ocean depth was about 30 ft. The island was to be connected to the mainland by a trestle causeway that would provide a two-lane access road and support for the water pipeline and the electric transmission cables.

The initial cost estimate for the proposed dual-purpose plant indicated a savings in site cost, water conveyance cost, and power-transmission cost relative to the best shore-based plant. For a variety of reasons, among which was rapid escalation in project cost estimates, the Bolsa project was not undertaken. To the reviewer's knowledge, no final study was made to determine whether the artificial island site would have remained the choice on the basis of the revised cost estimates.

Feasibility of Offshore Dual-Purpose Plants

A preliminary study was made at Oak Ridge National Laboratory (ORNL) by Arnold, Gall, and Morris of the feasibility of offshore dual-purpose plants.⁹ The reference site was in the southern California coastal region and was similar to the Bolsa Island site. The plant consisted of a 3500-MW(t) heavy-water-moderated organic-cooled reactor as the heat source for a dual-purpose plant producing 733 MW(e) (gross) of electricity and 250 Mgd of fresh water.

Under contract to ORNL, Ocean Science and Engineering Company, Inc., examined eight different offshore arrangements - five fixed-plant concepts and three floating-plant concepts. Three arrangements were chosen for evaluation:

1. A 10-acre artificial island with a caisson for reactor containment.
2. A deep-water caisson 330 ft in diameter, which would contain the entire plant and would be located in water up to 180 ft deep.
3. A floating station with a raftlike hull composed of five separately constructed prestressed-concrete units interconnected to form a single square hull 430 ft on each side. This concept is illustrated in Fig. 1.

It was concluded that all these concepts are technically feasible since similar offshore structures have already been built. Artificial islands have been built as far as 3 miles offshore and in water from 50 to 100 ft deep. Large concrete caissons have been constructed in deep water. Notable examples are the caissons constructed for the Delaware Memorial Bridge and the Verrazano Narrows Bridge. The Verrazano Narrows bridge has two large caissons, each of which is 230 by 130 ft and 130 ft deep. The concrete rafts

proposed as elements of the floating station are similar in size to concrete dry docks constructed and used (for over 10 years) by the U. S. Navy.

Estimates were prepared for the site-related costs for each of the proposed offshore plants, and these costs were then compared with those for an onshore plant in southern California. For this particular study, in which it was assumed that the offshore site was directly seaward of the onshore site, the island site was found to be the most economical choice if the sea were shallow. Among the offshore siting schemes, the floating station looks attractive for water depths of 100 ft or over since the costs of alternative offshore stations increase with water depth.

Floating Station

A study by Daniel, Mann, Johnson, and Mendenhall of a floating 1000-MW(e) nuclear power station was undertaken to determine the feasibility of providing earthquake resistance.¹⁰ The postulated site for the study was the same as for the proposed Bolsa Island station. A PWR similar to Indian Point 2 was assumed. Ten different barge designs were considered. A flat, rectangular, shallow-draft barge constructed of prestressed concrete was selected for detailed examination since it was the most economical within the constraints that it be within present (1968) technology and be suitable for shallow (30 to 40 ft) water application. The barge was tethered within a rock breakwater to protect it from waves. Other forms of wave protection were considered briefly but were not included in the reference design because further development would have been required to demonstrate their effectiveness. The main conclusions of the study were

1. The floating-station concept is an effective means of gaining earthquake protection for a nuclear plant that uses conventional light-water reactor technology, equipment, and arrangements.
2. The proposed floating station could be constructed with presently available techniques.
3. The cost of the floating station would be slightly greater than that of a similar station located on an artificial island. The artificial island would not, however, provide the same degree of earthquake protection. To do so would reduce or eliminate the differential cost.

Underwater Containment

The purpose of an ORNL study by Klepper and Bell was to determine the feasibility and potential

Fig. 1 Floating-platform dual-purpose nuclear power and desalination plant.

advantages of containing large light-water reactors in underwater caissons, with primary emphasis on reactor safety.^{11,12} The reference design consisted of a 715-MW(e) boiling-water reactor housed in a concrete caisson located $\frac{1}{2}$ mile offshore. The turbine-generator plant was located on a 10-acre artificial island, which surrounded the caisson. Although the site selected for the study was a hypothetical one, the characteristics were quite close to those for the proposed Bolsa Island plant.

Seawater was used for containment cooling and to reduce the outleakage of fission products in the event of a reactor loss-of-coolant accident. It was estimated that in the event of an accident the activity release rates to the environment would be several orders of magnitude smaller than for the corresponding above-ground containment system. The postaccident fission-

product-retention characteristics resulted from the fact that (1) the external sea pressure would oppose outleakage of fission products and (2) the caisson arrangement inherently forms a double-containment system that would provide significant fission-product holdup. Another important safety advantage of underwater containment would be the availability of reliable passive decay-heat removal with natural-circulation heat exchangers and heat dumped to the ocean.

Economics of Supplying Inland Load from Offshore Plants

A general study was made by Bowers and Wichner of the cost of supplying power to an inland city from an off-the-coast power plant relative to the cost of power from a near-load facility.¹³ It was assumed that

cooling water would be in short supply near the city and that a dry cooling tower would be needed for the near load plant; the tower would cause higher costs and result in lower thermal efficiency. In addition, the offshore site would have the advantage of requiring lower safeguards costs. On the debit side the cost of the offshore site (a floating station) would be higher, as would labor costs during construction.

The result of the study indicated that for a light-water reactor the cost of energy at the bus bar would be lower for the offshore plant. The savings in the cost of power generation were then applied to the cost of transmitting power to the inland city. It was found that, when plants of equal capacity were compared, transmission distances of up to 80 miles (including 10 miles of submarine cable) could be accommodated from the savings on generation costs on the basis of present-day transmission technology. When it was assumed that the large (10,000-MW) offshore plants would supply smaller (1000-MW) inland loads, allowable transmission distances ranged from 270 to 410 miles.

OFFSHORE PLANTS IN RELATION TO SITING PROBLEMS

The four broad problems mentioned earlier are responsible for the difficulty in finding acceptable sites for nuclear power plants. Offshore siting is discussed below in relation to each of these problems.

Cost and Land Availability

With respect to the availability of sites, there is no question that offshore siting would be advantageous because new "land" would be created. The necessity for competing with other interests for real estate would be obviated, and this could be an important factor in gaining public acceptance. Of the two offshore concepts, the floating station would seem to allow a wider choice of locations since it would be less affected by ocean depth or bottom contour. The possibility of relocating the station at some future time because of changes in population or consumption patterns might also be a factor in favor of the floating station.

The costs associated with siting depend almost exclusively on the location, so no general conclusions about costs can be made. If, as was the case for the Bolsa Island project, land near the load center is unavailable or if the land that is available is expensive, offshore plants could be the economic choice. The

relative costs of fixed and floating offshore plants depend on the depth of water and on the degree of earthquake resistance required. For shallow water and nominal earthquake resistance, the fixed offshore station could be more economical. For deeper water and greater protection from earthquakes, the floating station might be the choice.

One factor that could have a strong bearing on the cost of floating stations is the possibility of shop assembly in a permanent graving yard. The graving yard could be built at a coastal location with a favorable climate for year-round outdoor work to eliminate construction delays because of weather. Since the labor force would be permanently established, the labor cost would be less. More than one plant could be under construction at one time, and this would allow more efficient utilization of skilled labor. Large factory-assembled components could be delivered to the yard by barge, and this might allow the construction of larger, lower-cost plants. The completed plant would, of course, be towed to the final site. This should be a reasonably inexpensive operation since, even for a movement of 1000 miles, the time required should be less than a month.

The cost reductions that might be possible with floating stations have not been investigated but should be included in future work.

Environmental Effects of Plant Effluents

The capacity of the ocean to accept waste heat or low-level radioactive wastes is, of course, very large compared with that of the atmosphere or inland streams. The floating station appears to be especially attractive from the standpoint of heat rejection. The plant could be located in deep water, with the cooling-water intake drawing from the colder lower layer. The cooling water could be discharged at the surface near the normal temperature of the surface water to minimize thermal effects.

Probability and Consequences of Accidents

With respect to the probability of accidents initiated by plant malfunctions, there appears to be little, if any, difference between shore-based and offshore stations since essentially the same plant design and components would be used in both. The effects of natural phenomena could, however, be quite different. Clearly, offshore floating stations would be far less susceptible to earthquakes than either shore-based or

fixed offshore plants. However, floating stations, and to a lesser extent fixed offshore stations, would be more vulnerable to storms than shore-based plants.

The consequences of reactor accidents could be of less concern with offshore than with shore-based plants. First, there could be an important advantage in separation distance; the offshore plants, and especially floating plants, could provide a large exclusion zone in which there would be no permanent population. Second, either of the offshore-plant concepts could be designed to provide improvements, relative to land-based plants, in fission-product retention and post-accident cooling.

Visual Presence and Other Sensory Effects

Offshore stations could probably be seen from a longer distance and by more people than shore-based plants. On the other hand the impact of the visual impression would probably be less because the station would have its own setting, distinctly apart from other activities.

CONCLUSIONS

It may be concluded that many of the problems associated with siting nuclear power plants could be alleviated by the construction of offshore stations. There is no technical reason for not considering fixed (artificial island) offshore stations as a practical siting alternative at the present time. Of far greater future potential is the floating-station concept, since it offers the possibility of almost unlimited choices of offshore locations, of improvements in safety through earthquake resistance and large exclusion distance, and of cost reductions as a result of factory assembly. Considerable development will be required, however, before the potential benefits of floating stations can be realized.

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5.3 Underground Nuclear Plant Siting: A Technical and Safety Assessment

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Abstract: *The siting of nuclear plants underground is examined broadly on the basis of technical feasibility, design, safety, and economics. Discussion is general for all reactor types and underground siting alternatives, but specific details are mainly referred to a 1000-MW pressurized-water reactor located in a deep rock cavern. Conclusions are that current technology is entirely adequate and that the enhanced safety posture of an underground site may facilitate close-in siting and, with no greater risk or possibly less than that of a surface site. Economic trade-offs are identified, and it has not been clearly established that underground siting would result in an economic penalty.*

The concept of siting nuclear reactors underground is not new, as evidenced by the three experimental power reactors and one 275-MW(e) pressurized-water reactor (PWR) placed underground in Europe in the 1960s. Also, a 1200-MW(e) light-water reactor (LWR) plant planned for the Badische Anilin Soda Fabrik (BASF) at Ludwigshafen, Germany, is currently under active review after many postponements.¹ Numerous studies have been undertaken both abroad and in the United States on the feasibility of placing power plants of various types in underground locations for reasons of safety, economics, and/or waste-heat usage.^{2,3} The general consensus is that ample experience has been gained in underground hydroelectric projects and nuclear testing programs and that the technology exists for utilizing underground space for nuclear plants of any type.

Siting of large commercial nuclear plants underground has not proliferated for various reasons, such as (1) no real shortage of surface sites up to the early 1970s, (2) ambiguity of the licensing requirements, and (3) apparent economic penalty. However, the siting situation has changed owing to the current high cost of energy, the high demand for power, and the growing shortage of suitable sites. Underground siting has been considered as an alternative to surface siting and floating-station siting concepts. It offers the "depth

dimension" of the earth surface for beneficial utilization, particularly in areas where stringent values are attached to such criteria as population density, surface-land use, and as-low-as-practicable radioactivity releases.

Within the above context, it is an opportune time to examine the viability of the underground-siting concept as an alternative option for nuclear power-plant siting. This article summarizes key technical, safety, and economic factors associated with such a concept and neglects the broader questions of a philosophical, social, or socioeconomic nature.

UNDERGROUND-SITING ALTERNATIVES AND GEOTECHNICAL CONSIDERATIONS

The three variations of the underground-siting concept are as follows:

1. Surface mounded. The plant is essentially constructed at the ground level and the outside surfaces of vital structures are backfilled with soil and/or special material.

2. Cut and fill. The plant is constructed in an open-cut excavation and then covered with soil and/or special material. The cut-and-fill concept can be achieved either in soil or in rock.

3. Deep in rock. The plant is constructed in a cavern excavated at depth in a rock mass. The excavation can be achieved either horizontally into a mountain (or hill) or vertically deep below the earth's surface.

Within these primary variations, several arrangements of the plant layout can be effected. The plant may be completely underground or some portions may be on the surface, but the optimum arrangement will vary with area, depending on ground conditions and local factors. However, it would be of interest to note that all four underground nuclear reactors in Europe

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are placed within a hillside, with conventional turbine-generator facilities on the outside. Churchill Falls, the 5000-MW hydroelectric project and the largest of its kind underground, is sited approximately 1000 ft below the surface of the Labrador plateau. At first inspection, a deep-in-rock layout is more costly in terms of excavation but offers advantages in cavity stability and in earthquake-loading design for structures. For these reasons the following geotechnical discussion is largely based on the deep-in-rock concept.

It is fundamental that the rock cavities housing vital plant structures be stable even under uncommon circumstances. A considerable body of precedents compiled during the past 75 years of hydroelectric experience provides compelling evidence of the degree of stability achievable by large engineered rock caverns. For example, Cooke and Strassburger⁶ listed over 300 underground hydroelectric stations as of 1957. Since then, further advances have been made in underground engineering through such developments as the Churchill Falls installation and the Plowshare and weapons-testing programs.

Cavity span has been commonly cited as a possible difficulty because engineered rock cavities to date are often less than 100 ft in span, less than the diameter of a normal reactor containment building. The fact is that dimensions of underground hydroelectric plants have been sized to accommodate the necessary equipment and have not been limited by geotechnical constraints. Cavity spans of 110 ft have been successfully constructed at the Waldeck II pumped storage project in West Germany. Natural caverns with large spans exist, including the 600-ft span of the cavern at Carlsbad, N. M. It must be further noted that the cylindrical geometry of most containment buildings is well-suited to underground excavation.

Table 1 lists a few examples of natural and man-made caverns for both hydroelectric and nuclear plant siting.

Currently available methods for analyzing underground caverns include finite-element and finite-difference techniques that are capable of handling the design of cavities in rock for static loading conditions.⁸ These techniques have been refined for analyzing dynamic loading conditions, but their use has not been extended to a complex arrangement of rock cavities.⁹ In recent years a great deal of experience and predictive capability was gained by the University of California, E. O. Lawrence Radiation Laboratory at Liver-

more, the Sandia Corporation, and others who participated in the underground testing of nuclear devices.¹⁰ This capability includes the engineering of underground cavities to withstand intense blasts, the closure and isolation of these cavities to control the release of fission gases, and the containment of fission products in accordance with characteristics of soils, rocks, and groundwater. Certain aspects of earthquake-response spectra can also be simulated by nuclear blasts.

There appears to be no need for new technology either for excavation or for support of rock caverns. Conventional drilling and blasting techniques are adequate to allow excavation of the requisite cavities within the time scale appropriate to the overall construction schedule. Currently available support techniques include stressed rock bolts and tendons, wire mesh, pneumatically applied concrete, and, occasionally, mass or reinforced concrete with structural steel. In general, modern design techniques utilize the rock as a supporting medium around the cavities and do not depend on such passive means of support as concrete or structural steel.

Preconstruction-site investigation must be extensive in order to establish early design data, such as the nature of the bedrock, faulting and discontinuities, and groundwater characteristics. The use of exploratory shafts will be more extensive than in the case of site exploration for surface plants.

PLANT DESIGN CONSIDERATIONS FOR UNDERGROUND SITING

Current commercial power plants in the United States are of the boiling-water-reactor (BWR) type, the pressurized-water-reactor (PWR) type, or the high-temperature gas-cooled reactor (HTGR) type. Future reactor concepts include the liquid-metal fast breeder reactor (LMFBR), the gas-cooled breeder reactor (GCBR), the molten-salt reactor (MSR), and the HTGR coupled with the gas turbine. In other countries, heavy-water reactors (HWRs) and advanced gas-cooled reactors (AGRs) are in use for commercial power generation.

There appear to be no major constraints to siting any of the above reactor plant types underground. In fact, existing underground facilities already include two boiling heavy-water reactors (Halden, Norway) and

Table 1 Examples of Existing Rock Caverns

Name and location	Principal dimensions	Rock type	Remarks
Natural Caverns			
Natural Tunnel, Va.	130-ft span, 75 ft high, 900 ft long		Used as rail tunnel
Mammoth Dome, Ky.	150-ft span, 250 ft high, 400 ft long	Limestone	
Rainbow Natural Bridge, southern Utah	270-ft span, 305 ft high	Sandstone	Arch
Carlsbad Caverns, N. Mex.	625-ft span, 300 ft high	Limestone	
Engineered Hydroelectric Caverns			
Portina, Tasmania	45-ft span, 84 ft high, 300 ft long	Madstone, shale	Rock bolts; mesh and gunite lining; 500-ft rock cover
Marrow Point, Colo.	57-ft span, 134 ft high, 206 ft long	Schist, quartzite, granite	400-ft rock cover (1968)
Portage Dam, British Columbia	67-ft span, 144 ft high, 890 ft long	Sandstone, shale	Poor rock condition; concrete roof arch provided; 200-ft rock cover (1970);
Churchill Falls, Labrador, Canada	81-ft span, 145 ft high, 1000 ft long (main cavern)	Granite, granite/gneiss	Largest hydroelectric project in Western World (5000 MW); 1000-ft rock cover; 2.5- by 10 ⁶ -yd ³ excavation (1971)
Serca-Moneno, Italy	95-ft span, 92 ft high, 633 ft long		
Waldeck II, West Germany	110-ft span		Pumped storage facility
Engineered Test/Defense Caverns			
Norad Facility, Colorado Springs, Colo.	42-ft span, 60 ft high, 600 ft long	Granite	Rock bolts and mesh; three chambers (1964)
Nevada Test Site	60-70-ft hemisphere, 35-ft hemisphere	Tuff, Granite	Red-hot and deep-well test series Rock bolts and mesh; survived nearby blasts
Engineered Nuclear Plant Caverns			
Halden, Norway	33-ft span, 85 ft high, 98 ft long	Gneiss	Rock cover 100 to 200 ft; lined with reinforced concrete and painted (1960); 20-MW(t) boiling heavy-water reactor
Agesta, Sweden	54-ft span, 130 ft high, 175 ft long	Gneiss/granite	Rock cover 56 ft; lined with concrete and steel (1964); 60-MW(t) boiling heavy-water reactor
SENA, Chooz, France	61-ft span, 134 ft high, 140 ft long	Chalk/shale	Grouting: 1/2-in. steel lining; 275-MW(e) PWR
Lucens, Switzerland	59-ft diameter, 98 ft high	Sedimentary molasse	Two layers of concrete; aluminum foil and bitumen seal; 8.3-MW(e) gas-cooled heavy-water reactor; partial meltdown 1969; decommissioned; cavern destined for radwaste disposal

Agesta, Sweden), a gas-cooled heavy-water reactor (decommissioned, Lucens, Switzerland), and a PWR [Chooz-lez-Givet, France (SENA)].⁶ Although these plants are of small or moderate sizes, feasibility studies have been made^{2,4,8} for reactors with capacities ranging from 500 to 1000 MW(e). These studies have covered various possible layouts of the installation and have identified no major constraints that would preclude siting any particular reactor type underground.

In general, underground siting is most logical for a compact nuclear plant whose safety features are enhanced by its underground location. Candidates for this category of plants include the breeder reactor and the HTGR closed-cycle gas-turbine plant, particularly when they are coupled with dry cooling towers. However, due to the inherent incompatibility of water and sodium, special considerations will be required in the design of the cooling system of LMFBRs to prevent flooding of the underground cavity.

As a basis for amendment, a 1000-MW PWR plant, similar to Indian Point 3, was selected for the reference design. Several engineering modifications, either necessary or preferable, are discussed below.

Engineered Safety Features

The basic safety-design philosophy of surface PWR plants is retained on the basis of past experience and projected licensing requirements. Thus the plant shall have engineered safety features, such as containment, emergency core cooling, long-term heat-removal capability, and devices to control the effects of accidents. The underground containment does not have to be a duplicate of a surface containment to perform the same function. Its free volume can be reduced by a factor of 2, just enough to accommodate the service equipment inside it. This is because the containment volume has traditionally been sized to withstand the loss-of-coolant accident (LOCA) loadings. For a surface plant the most economic containment is one which is designed for a maximum pressure of around 50 psig and a temperature of 300°F. When the containment is built inside a rock cavern, it can be supported from all sides and consequently can withstand a higher pressure and temperature transient without violating the integrity criterion. For the same reason the containment liner does not have to be $\frac{1}{2}$ in. thick, because it only performs the leaktightness function, but the surrounding rock would take all the LOCA pressure loadings.

(The SENA containment liner is only $\frac{1}{4}$ in. thick, whereas the Indian Point 3 liner is $\frac{1}{2}$ in. thick at the cylindrical portion and $\frac{1}{4}$ in. thick at the hemispherical portion.) Furthermore, the 11-ft-thick foundation mat of surface containment, which consists of concrete with steel reinforcement, can be reduced because the bedrock (approximately 30,000 psi) provides an excellent foundation. All these factors combine to make the design and construction of the containment faster and cheaper and thus tend to offset disadvantages that excavation would cause to the construction schedule and investment cost.

The change of the containment volume will necessitate some changes in other engineered safety features, such as the hydrogen-control system, the ventilation and filtration system, and the containment sprays. However, these changes will be generic, and the details will not be covered here.

Other engineered-safety-feature devices unique to underground siting include (1) the sealing and isolation system, (2) the emergency flooding system, and (3) the emergency exit system. Items 1 and 2 are to ensure the additional containment capability of the underground site even in the case of a Class 4 accident (an accident more severe than the design-basis accident), and item 3 is to ensure the safety of in-plant workers in the case of emergency.

Additional engineered safeguards that can be provided for the underground site without adding significantly to its cost include the use of construction cavities for radioactive-gas storage and the use of rock saturation techniques to enhance the containment capability of rock.

Earthquake Design for Structures

Since an earthquake is the most outstanding common-mode initiator of accidents, earthquake-resistant design has been the major determining factor in plant siting and economics. Many sites have been discarded on the basis of questionable soundness of the local ground, and other sites, such as San Onofre 2 and 3, have been subjected to a very costly aseismic design. Designing plants to withstand earthquakes is expected to be less of a problem for good rock underground sites than for surface sites.

It has been a common experience after strong earthquakes that, within the same locality, even similar structures suffer unequal damages. Structures founded on or in rock usually suffer the least, whereas those on loose soil suffer the most. Many factors are responsible for this behavior, including the fact that soil amplifies

⁶SENA, Société d'Énergie Nucléaire Franco Belge des Ardennes.

earthquake motions because of its low density and elastic modulus. The effect of ground liquefaction and subsidence often accounts for creating foundation failure and/or landslide.

There have been convincing indications that earthquake loadings are substantially less on structures located in deep bedrock than on surface structures. The effects of earthquakes, both overseas and in the United States, have been studied,^{11,20} and many measurements of small seismic waves and man-made explosions have been made (Table 2). The general consensus is that, for the same earthquake and at the same geographical location, ground motion is less in bedrock than on the surface. The reduction factor is period-dependent, ranging from more than 4 for periods of around 0.3 sec to about 1.1 for periods of around 4.5 sec. The response spectra are also different for a bedrock medium. It is expected that the modified earthquake-response spectra would have a substantial design impact and cost reduction for the underground plant. Table 2 lists some documented evidence of lower seismic loadings on structures built on or in bedrock.

In addition to the reactor containment, other principal plant structures are the control room, the turbine-generator building, the emergency diesel-generator hall, the transformer gallery, and the rad-waste building. In many cases the rock cavities can be directly utilized as the housing structures for equipment, thus eliminating conventional structures. The rock walls can be further utilized for support and shielding, which are conventionally provided by reinforced concrete. Other design concerns for underground siting include provision for drainage of groundwater (high pressure, low volume leakage) and for corrosion effects of water on structural material. Figures 1 and 2 are vertical and plan sections of a conceptual underground-plant layout.

Equipment

The three main considerations concerning equipment of an underground plant are (1) sizes and weights of major pieces of equipment and their transportation to and from the underground cavities, (2) the erection and support of the equipment, and (3) equipment unique for the underground location.

Major items of equipment include the reactor pressure vessel, the steam generators, and the turbine generators. Since these items are built to strict standards, it is preferable that existing designs be used to the largest extent possible. The reactor pressure vessel, which could measure up to 22 ft in diameter and weigh

up to 400 tons (i.e., BWR reactor vessel), is often the heaviest item of equipment. It can be prefabricated and transported to the cavern or can be assembled in situ. The underground access tunnel can be as large as 30 ft in diameter and constructed at about 7° grade. Although the transport of the vessel through this adit would temporarily disrupt traffic, this is the matter of detailed scheduling. If site fabrication of the reactor vessel is preferred, the controlled environment of the cavern may prove to be an asset. The transport and erection of the reactor vessel of reactor types other than BWRs would be less difficult.

The ability to restrain piping and heavy equipment from all directions is a definite asset for the underground site. The cost of support and the likelihood of vibration failure and pipe whip are reduced.

Electric power produced by the generators at 10 to 15 kV can be transmitted to the surface either at generator voltage or at higher voltage after transformation underground. Hydroelectric practice provides precedent for either alternative. At Churchill Falls, Canada, power is transformed from 16 kV underground and then transmitted 900 ft vertically to the surface through 230-kV 500-MVA oil-filled cable. At Boundary, 150 ft of 13.8-kV isolated-phase bus connects the generators to the transformers, which are located in open-sided caverns at the surface, and, at Chutes des Passes, power is transmitted at 13.8 kV by an open aluminum bus 400 ft to the surface. Some development work is required on the sealing of high-voltage cables, but current technology is adequate for the sealing of voltages up to 22 kV.

For a location below the surface, the cooling-water head of some 500 ft requires that the condensers be built to withstand high pressures. Reliable isolation valves must be installed in the water conduits to prevent accidental flooding. The design of this type of equipment is well within current technology. The condensers of nuclear submarines have been designed to take deep dives. Current large condenser designs include the thick-tubed condensers of pressure capability above 100 psi for operation in conjunction with dry cooling towers.²¹

SAFETY CONSIDERATIONS

Would underground siting result in a significant improvement in safety? From the viewpoint of design, construction, and operation, it is clearly impossible to have a safe plant underground if it is not safe on the surface in the first place. However, placing the nuclear plant underground may provide some additional levels

Table 2 Evidence of Lessened Earthquake Loading in Deep Bedrock

Year	Author	Earthquake or data source	Pertinent conclusions on acceleration amplitude, displacement, velocity, or intensity ^a	Ref.
Foreign				
1930	Inamura	Kanagawa (1923) Kyoto (1930), 34-station records	Acceleration in tertiary rock, 0.1 g in sandy soil, 0.5 g Seismic intensity in hard rock 4 to 5 times less than in soft soil	
1931	Nam	Tokyo (1923)	Amplification in tunnel 1.1 to 4.1 less than on surface, depending on period of motion (4.5 to 0.3 sec)	11
1934	Inouye	Comparative study	Amplification in deep locations 1.1 to 3.10 less than on surface, depending on period (0.8 to 0.2 sec)	12
1934	Saita and Suzuki	Comparative study, Tokyo data	Acceleration at 68-ft depth $\frac{1}{2}$ to $\frac{1}{3}$ that at surface	13
1951	Kami and Tanaka	Iberagi mines	Underground seismic motion 1 to 5 times less than at surface	14, 15
1953	Kami et al.	Matsumoto area	Velocity spectrum in bedrock flat; empirical formulas for bedrock motion proposed	16, 17
1960	Morahan and Yabuki	Strong earthquakes	Intensities in alluvium in hilly region about 4 times less than in other areas	
1962	Tandon	Anam (1962)	Intensities in bedrock foundations far less than in soft soil foundations	18
1962	Tandon	Anat (1956), Bhuj town	Eastern portion of town founded on alluvium heavily damaged; western portion founded in rock suffered little damage	18
1962	Shirai	Tokyo seismic motion	Amplitudes at surface always greater than those underground	19
1966	Takano and Hagiwara	Nokogiriyama and Tokyo microseismic noise, up to 380 m deep	Seismic noise 2 to 10 times less in underground locations	20
1973	Okamoto	Book	Damages for tunnels in soil can be 1 to 4 times larger than in hard rock; thickness of lining cannot prevent damage	21
United States				
1954	Newman	Nepal-Bihar (1934), Kangra (1905)	Soft soil amplifies seismic motion; damage of structures built on hard bedrock less than on soft soil	22
1954	Steinbrugge and Morae	Southern California	Factual study of damages to oil wells, railroad tunnels, and earth dams; no clear comparative conclusions	23
1954	Stenberg	Pasadena seismic motion	Amplitude of motion in bedrock up to 10 times less than in poor soil	24
1959	Carder and Cloud	Motion due to explosion	Amplitude recorded in alluvium about an order of magnitude higher than calculated for hard rock	25
1959	Duke and Leeds		Damage by nuclear blasts and earthquakes to well constructed underground structure less than to surface structure	
1969	Kuempel and Azuks	San Francisco BART system design	Design earthquake has peak horizontal acceleration of 0.5 g in thick alluvium and 0.33 g in areas with alluvium less than 7 ft thick	26, 27
1972	Leeds	Sonora, Ariz. (1887), Montana (1925), Kern Co., California (1952)	Effect underground in the mines much less than that on the surface and diminishes with depth	28

^aAcceleration (measured in g) determines the inertial force that the structure is subject to and is normally the most important design parameter. Amplitude is the magnitude of the seismic wave as drawn by the seismograph (Richter scale). Displacement is the magnitude of the physical departure from an original reference frame, ordinarily the only quantity that can be physically measured. Velocity is the product of the displacement and the natural wave frequency. Intensity is a relative judgmental measure of the effects of earthquake (Modified Mercalli scale).

Fig. 1 Conceptual underground layout of a PWR plant, vertical section.

Fig. 2 Conceptual underground layout of a PWR plant, plan section.

of safety in specific circumstances. It is important that safety considerations be weighed within the framework of current safety-design criteria and that trade-offs be made among economic, environmental, and social factors concurrently with the maintenance of a common standard of risk.

General Safety-Design Requirements

The stored fission products in the core of a 1000-MW(e) reactor (of the order of 10^9 Ci), if released to the atmosphere, would cause tremendous loss of lives, properties, and long-lasting effects.³⁰⁻³² The objectives of safety engineering are to assure the health of the public in the event of inadvertent releases of fission products to the ecosystem and also to provide similar protection for the plant operators.

Events that can give rise to the release of fission products have been extensively discussed in the licensing process of nuclear plants, summaries of which can be found in the U.S. Atomic Energy Commission (AEC) regulatory safety guides and in Report WASH-1250.³² In general, plant structures and equipment must be designed to withstand weather-related phenomena (tornadoes, hurricanes, floods, snow, etc.), earthquakes, tsunamis, and missile and aircraft impacts. Protection must also be provided against such in-plant phenomena as power excursions, fuel-handling incidents, rotary machinery failures, and pipe breaks. Of the nine classes of accidents categorized by the AEC Regulatory Staff, Class 8, the double-ended break in

the largest primary-coolant pipe, has been termed the design-basis accident (DBA) because it results in the largest credible fission-product release from the core and the highest thermal-hydraulic loadings on the containment system. Probability for the DBA is currently assessed to be less than 10^{-4} per year per system.

Current-generation nuclear plants are not designed to withstand Class 9 accidents, which have an extremely low probability of occurring but which involve a larger release of fission products than the DBA. Events or series of events that can lead to a Class 9 accident include the following:

1. A sudden large rupture of the reactor pressure vessel and loss of coolable core geometry.
2. Anticipated transients without scram and subsequent failure of protective devices.
3. Failure to provide emergency power, emergency core-cooling injection, long-term heat removal, and/or failure of containment integrity following the DBA or lesser accidents.
4. Sophisticated sabotage that can lead to any of the above.

It is generally believed that, if such a low-probability event should occur, the damage to the public and to the environment would be severe even if emergency evacuation plans were effectively carried out. A basic rationale for the current practice of not siting nuclear plants in metropolitan areas is the extra precaution taken in light of these low-probability events.³³ If it can be shown beyond doubt that

protection can be reasonably provided for Class 9 accidents, the above precautionary measure may certainly be relaxed. An underground site appears to hold such capability, but much needs to be done in terms of concrete analytical and experimental evidence. Only a positive evaluation program concerning a real design can give such definitive evidence. The advantages and disadvantages as assessed at the present state of the art are discussed below.

Safety Concerns Associated with Underground Siting

Most safety concerns for the underground siting of a nuclear power plant can be classified into three categories: acceptance factor, engineering problems, and groundwater contamination.

Acceptance Factor. To most technical people and to the common public, the placement of a nuclear plant underground is a definite advantage. The concern of some is that there would be unforeseen problems, for example, inspection and operation would be more difficult. The statement that "underground siting would provide additional safety" can be misunderstood as an inference that surface plants are unsafe. The fact is that surface plants have been designed to withstand all the accidents that are considered credible during their lifetime. Therefore underground siting would provide an alternative to surface siting but would not replace it.

It has been noted that the population in the vicinity of nuclear plants has increased faster than ordinarily projected, which indicates the readiness to accept a small risk in exchange for such other benefits as a real-estate-tax reduction. When the level of risk of an underground plant has been demonstrated and accepted as being small, siting close to load centers could be considered.

Current AEC and industry programs on standardization, quality assurance and control, in-service inspection, and reliability design apply to all types of nuclear power plants. It is expected that the siting of a plant underground will be based mostly on the strategy of optimal choice, and the public acceptance would certainly be a major determinant of this choice.

Engineering Problems. Like the off-shore siting concept, the idea of placing a nuclear plant underground also has its related engineering problems. These include the design of piping runs, access ports, high-pressure condensers, drainage system, ventilation system, and redundant heat sink. At present, little has been done to provide a definitive solution, although it

does not appear to be beyond the state of current technology.

Conduits to carry the circulating water must run from the surface to the underground cavity unless a large underground water reservoir is provided. It has been suggested that the design of these conduits to withstand earth displacement would be difficult and thus could negate the argument of reduced seismicity in rock cavities. However, a nuclear plant would not be placed close to a fault, which would be discovered during the extensive site investigation. Hundreds of miles of water conduits have been built underground for hydroelectric stations and water-supply systems in areas of high seismicity, and no major problems were encountered. Further, these circulating-water conduits do not contain fission products and can be placed in well-sealed tunnels and thus receive the benefits of reduced seismicity. For conventional surface plants the circulating-water conduits are often of considerable length, often constructed in sediments and potentially exposed to seismic loading.

As discussed in the section on plant design considerations, the use of high-pressure condensers is necessary for a plant deep below the surface. The accidental flooding of the cavities from a break in the circulating system (caused by missiles, explosions, flywheels or earthquakes, etc.) is a new category of accidents for which the plant must be designed. The provision of an isolation-valve system in the water conduits and a high-capacity flood drainage system is within the state of technology and economy. The current design of the residual-heat-removal system should be able to provide core cooling in the event of a problem in the circulating-water system.

Engineering problems involving groundwater leakage inevitably arise at levels of 300 to 500 ft below the surface. Although the pressure of such water is theoretically high, the permeability of most rock (10^{-5} to 10^{-7} cm/sec) is so low that the leakage would also be low. For example, at the Lucens cavern, which has a volume of 7600 m³, a surface area of 2100 m², and a 40-m rock cover, the water-inseepage rate is 5 m³/day and is not changed by weather or season.³⁴ The design of structural walls and/or grouting must allow for a porous layer from which the groundwater inseepage can be drained. From the safety viewpoint, care must be taken to protect structures and equipment and their supports from the effects of water corrosion, which may breach their designed safety functions.

The ventilation requirements of an underground plant must obviously be higher than those for a plant on the surface, where natural draft can be utilized for

most of the structures except the containment. High-capacity and redundant ventilation systems will be needed to keep the in-plant environment safe and the radiation level as low as practicable. In the case of an accident, emergency ventilation must also be available. Many years ago, Von Ubiach³⁵ proposed a pressurized-air flushing system to assure ventilation under any circumstances.

Capability of the underground-facility design to prevent leaks to the surface (at ground level) from the multiple accesses and penetrations has been frequently questioned. There is doubt that leaktightness can be achieved in the case of accidents. The question is how tight should seals be to achieve the same level of containment as with surface plants. Proponents point out inherent systems, such as the large usable volume of the tunnels, the normally closed multiple valves and locks, the normally lower-than-atmospheric cavity pressure, and sealing by flooding. Experience with the underground nuclear testing and the Plowshare program shows that the technology is sufficiently advanced to provide leakfree penetrations and valves at cavity pressures far exceeding those which can ever be expected from a commercial LWR plant.³⁶

On the basis of current technology, definitive answers to the questions of ventilation, monitoring, and sealing can be obtained with only moderate engineering and analytic efforts.

Groundwater Contamination. Perhaps the most common serious reservation about the prospect for underground nuclear siting is the possible long-term contamination of groundwater. The concern for water contamination is not a problem for underground plants alone, however. Under normal plant operation, both surface and underground plants have restrictions on the release to the drainage system and service water. Under severe accident conditions involving core meltdown, the contamination threat to groundwater appears to be reduced for an underground site.

There has been over 25 years of experience with the storage of radioactive wastes in underground tanks. At Hanford, large volumes of low-level wastes have been released to sedimentary soil, and the local groundwater has been routinely monitored in hundreds of wells. Tritium, ⁹⁰Sr, ¹³⁷Cs, and nitrate ions were detected at low concentrations as far as 4 miles distant. Other nuclides of more safety significance were only found in traces and close to the area of injection. Furthermore, water samples from very deep aquifers have not contained fission products above background levels.³⁶

Experience with the underground nuclear tests indicates that tritium is found only in minor levels, even in probing wells drilled close to the chimney. Monitoring of water in the vicinity of the shaft showed no traces of fission products.³⁶

Current radwaste facilities at nuclear plant sites are designed so that a hypothetical release coupled with flooding will not result in groundwater doses at the site boundary in excess of values given in 10 CFR 20. It is unlikely and illogical that an underground plant should be designed with a different set of criteria.

In the case of very severe accidents, the underground site provides better protection for both surface water and groundwater for the following reasons:

1. The underground site is selected to be in competent rock with very low permeability. It takes days, even years, for water to travel a short distance. Fission products (except tritium), owing to ion exchange and other phenomena, travel at least an order of magnitude slower than water.³⁷⁻³⁹

2. The driving force resulting from a LOCA is quickly neutralized by the rock porosity and the piezometric pressure of groundwater. It has been estimated that the volume of the containment is practically doubled because the porous space of the rock is only about one radius thick outside the containment walls.⁵ Furthermore, industrial practice for many years has been to store gasoline, oil, natural gas, and even helium in deep unlined caverns so long as the designed overpressure is lower than the local piezometric pressure.

3. Engineering safeguards can be included, such as the use of special materials, like clay or grout, to further reduce the permeability of rock.^{5,40,41}

4. The preferential path for the diffusion of fission products from the containment structure is not into the rock and groundwater, but toward other cavities of the plant.

Inherent Safety Features of an Underground Site

Immunity to Surface Phenomena. It is obvious that an underground site is not subject to the natural and man-made hazards of a surface site. This is true for portions of the plant that would be most logically placed underground, such as the primary containment and the spent-fuel-storage pool. Their design is normally dictated by such requirements as the DBA and earthquakes and is inherently resistant to surface phenomena. Other plant structures (such as the water intake and outfall, the cooling towers, and possibly the turbine-generator building and the switchyard) may be

at the surface and therefore are not immune to surface phenomena.

Where stringent requirements to withstand air crashes and explosions are imposed, considerable economies appear to be achievable with primary portions of the plant placed underground. Also significant is the inherent capability of underground structures to withstand acts of war. Earlier studies of the Department of Defense and the AEC gave due regard to such features.^{1,42}

Modified and Lessened Seismic Loadings. This topic has been discussed in previous sections, including some recent evidence of the lessened seismic loadings on structures in bedrock cavities. The evidence of modified seismic spectra in bedrock and the resulting reduced inertial force loading on structures within rock cavities needs further confirmatory studies. Earthquake-resistant design for the plant structures and equipment could be determined more accurately and more economically owing to modified and reduced loadings and to the ability to support at all elevations. The probability of accident (e.g., a double-ended pipe break) and/or breach of the containment is therefore reduced. However, in addition to in-plant emergency safeguards similar to those for surface plant designs, an underground containment possesses further overpressure and fission-product retention capabilities. From a safety standpoint this feature of simultaneous reduction of accident-occurrence probability, although increasing the capability to mitigate the consequences, is especially attractive.

Additional Level of Containment. The thickness of the rock overburden above the cavity represents a formidable barrier against the release of radioactivity to the atmosphere following an accident. Of course, rock, like concrete, is not permeable, but a mental image of this additional level of containment of fission products can be made by imagining a surface nuclear plant with a containment building having a wall thickness of several hundred feet.

The containment capability of rock has been investigated by many authors.^{2,34,43} Most studies were in connection with the measurement of rock porosity and permeability necessary for the containment design of the Halden, Agesta, SENA, Lucens, and planned projects. Quite significant was the experimental work done at the Rävfall Mountain in Sweden and at the Halden site. In the Rävfall tests, Bernell and Lindbo⁴³ measured the air-leakage rate from pressurized-test rock chambers using pressure devices and radiotracer techniques. They found that (1) for a fixed overpres-

sure, the leak rate quickly reached an asymptotic value; (2) most of the leakage was from the top of the chamber, indicating that the direction of leakage is toward the lower pressure sink; (3) when the pressure in the chamber was suddenly reduced, air from the rock flowed back to the chamber; and (4) the leak rate was considerably reduced when the rock was saturated with water.

In 1971 the Institute for Atomic Energy in Norway, in cooperation with the Norwegian Geotechnical Institute and the Norwegian Water Resources and Electricity Board, carried out a series of experiments at the Halden site using a method similar to that of Bernell and Lindbo. Results showed that the leakage of air from the cavern was predictable on the basis of rock permeability and that the filtration of the radiotracers was uneven for different isotopes and at different measurement stations.²

A comparison has been made between the release from a surface containment and from an underground site. The surface containment was assumed to be either of the Indian Point 3 type or of the double-containment type, and the underground containment was assumed to include a porous concrete layer between the liner and the rock. The delay time of release from the ground surface was used as a parametric factor, thus consolidating all the currently unknown mechanics of transport and retention in rock. Some analyses indicate that a reduction in the accident release by as much as a factor of 100 might be achieved by the underground containment.⁴⁴

However, such a parametric study is not definitive. The phenomenology of release from the underground site cannot be correctly described in the absence of a specific site and design. Figure 3 is a schematic of different routes and principal processes invoked. In the event of a severe LOCA, the release to the containment is limited by safeguards, such as the emergency core-cooling system and the residual-heat-removal system. The fission products are cleaned up from the containment atmosphere by the containment sprays, the recirculation cleanup system, and by plateout. The containment also has a hydrogen-control system to prevent dangerous buildup of hydrogen concentration. Any leakage through the liner would be collected in the porous layer of concrete between the liner and the cavern walls. From that location the gas, steam, and/or fission products either would go to the drainage system and other cavities or would penetrate into the rock pores. After a short distance into the rock, the pressure would be asymptotic to rock water pressure. The flow would be reversed when the driving leakage pressure

Fig. 3 Phenomenology of fission-product release from an underground site following a LOCA.

had been suppressed. From this schematic it is clear that the release of fission products to the atmosphere would be possible either by permeation through rock when there is no letup of the LOCA pressure or by leaking through the multiple seals in the penetrations.

The likelihood of both possibilities seems to be small because the pressure-driving force is quickly neutralized by pressure suppression and the inherent volumes in the cavities and in rock pores. Furthermore, the engineered safeguards (i.e., multiple locks, positive-pressure seals, and rock saturation) provide additional barriers.

Class 9 Accidents. As discussed earlier, Class 9 accidents are low-probability events that can happen only under a series of most unlikely mishaps. However, major policy decisions of fundamental importance to the nuclear industry have been based on such hypothetical accidents. These include the Price-Anderson Act, the criteria on population density, exclusion distance, and the low-population zone.

The scenarios of the release of fission products following a Class 9 accident with core meltdown have

been discussed by various authors.^{30,45,46} Subsequent to the core meltdown, the phenomena of pressure buildup, carbon dioxide generation, steam explosion, hydrogen explosion, etc., will almost certainly breach the primary reactor containment. Volatile fission products would be released in a puff, and other fission products would follow as a function of time. Within a few days after the accident, tens of miles of land surrounding the plant would certainly be severely irradiated. In spite of any evacuation measure, damages in terms of lives and properties would be very high.

Current nuclear plants are not designed to cope with Class 9 accidents, because of the unreasonably high cost of such a design to cope with a highly unlikely event. However, no large nuclear plant to date has been placed in a metropolitan area, since it is desirable to take advantage of distance and meteorological factors. It can be argued that an underground site utilizes the rock overburden and other engineered safeguards to obtain the same benefit or to improve on the benefits that can be provided by distance and meteorology. Furthermore, features unique to the underground site include assurance of long-term heat removal by ultimate flooding and by delay of any release long enough for such emergency measures as evacuation of the population to be in full effect.

The Balanced View on Safety

The above discussion on the safety merits of underground nuclear siting indicates that there is no overwhelming advantage nor disadvantage that can make or break the concept. The current consensus in safety circles is to view all matters of safety in a balanced way. When the safety question is of the absolute sense (such as a Class 9 accident), it is put in the perspective of history and human experience. When the question is a matter of comparison between competitive approaches, the relative risks and benefits are evaluated.

Current site-selection activities include such factors as safety, environmental impact, cost, and need. The evaluation of these factors is complex but nevertheless imperative in the modern society when the use of natural resources invariably brings about tangible and intangible impacts on nature and society. The recent California siting study is an example of a coordinated, balanced evaluation of different power-plant-siting concepts to satisfy California's power need in the near future.⁴⁷ In this study, four surface siting concepts, four underground siting concepts, and seven off-shore concepts were evaluated according to cost, environ-

mental impact, feasibility, and risk. An overall acceptability figure of merit was defined based on factors that could be clearly delineated. The study concluded that, in California, surface sites are more competitive than other concepts, the underground siting being constrained by a shortage of suitable sites.

The concept of risk has been found to delineate well the safety aspects of a nuclear plant.^{48,49} Questions relating to fission-product release, groundwater contaminations, sabotage, etc., can be placed on a common perspective and their risk contribution quantified using failure modes and effects analysis and/or logic-tree techniques.⁵⁰⁻⁵²

From a safety viewpoint an underground site should be evaluated in conjunction with other surface and off-shore sites, in that it should impose only a small and acceptable risk to society.

ENVIRONMENTAL CONSIDERATIONS

Environmental-impact considerations are taking an increasingly important role in the choice of site and type of plant. These considerations include the questions of water use, land use, air pollution, chemical and fission-product discharges, and effects on the ecosystems.

Short of a specific site and plant, it would be general at best to discuss the advantages and disadvantages of underground siting. Some work stresses the aesthetics of an underground plant because it does not protrude into the landscape. This is true if the total plant is to be underground, but design considerations indicate that some major structures must still be on the surface (e.g., the switchyard, the cooling towers, and, in some cases, the turbine-generator building).

Use of the depth dimension of land for the benefit of socioeconomic activities is certainly advantageous in view of projected human population and needs. If a plant sited underground could be located close to load centers, many benefits could be realized, such as the following:

1. The exclusion distance would be considerably reduced, and the surface land directly above the plant site could be used for recreational or other purposes.
2. The need for large tracts of land to put up transmission lines would be reduced.
3. The savings in transmission power loss could result in a higher MW(e)/MW(t) ratio and therefore conserve fuel.
4. The proximity to economic centers could encourage the effective use of waste heat, thus further enhancing the conservation of fuel resources and

further protecting the environment from the impact of thermal effects.

In the event of a very severe accident, the underground site may protect the air, water, land, and food from pollution, and disruption to other economic activities would be less drastic. As discussed previously, long-term groundwater contamination is less likely for an engineered underground containment.

ECONOMICS

The economics of an underground site compared with a surface site seems to be the only major factor that would determine the viability of the underground concept. A review of the limited cost data for the four European underground reactors indicates a slightly higher cost than if they were built completely on the surface. Most European and United States feasibility studies estimate an increase between 5 and 10% of the direct cost of a 1000-MW(e) nuclear plant (Table 3).

Since there has never been a nuclear plant of the 1000-MW category built underground, it would be presumptuous to claim accuracy to any estimate. Experience indicates that estimates for conceptual, first-of-a-kind, industrial plants have always been low because of many unforeseen problems.

The economic penalty for the underground plant includes (1) cost of excavation; (2) cost of engineering systems, such as the drainage system, ventilation system, high-pressure condensers, and isolation system; and (3) scheduling and logistics. Potential savings include the following:

1. Savings in transmission right-of-way.
2. Savings in transmission loss (which is considerable over the lifetime of the plant).
3. Savings in containment design and construction (less volume, less steel, and less concrete).
4. Savings in structures and equipment support.
5. Plant construction time shortened (not including caverns) owing to the controlled environment.
6. Potential use of waste heat.

Excavation is often mentioned as the single most important factor in the cost increase. On the basis of the Churchill Falls experience, Table 4 shows a cost of approximately \$21 million for the excavation of 770,000 yd³ of rock necessary to place a plant similar to Indian Point 3 underground. This represents 10% of the plant's direct cost, which must be disbursed before construction of the actual structure is started.

Schedule is another factor. It has been estimated that an underground plant will take 2½ years longer to

Table 3 Cost-Increase Estimates for a 1000-MW(e)
Underground Plant Vs. a Surface Plant

Author and/or affiliation	Ref.	Cost increase
C. Beck, AEC	3	3 to 7%
P. Erkes and P. Grau, SENA	53	More expensive than if built on surface; \$500,000 for 36,000-m ³ excavation alone
F. C. Rogers, Harza Engineering	4	\$10 per kilowatt
W. P. Swiger	54	Rock excavation only part of cost in- crease; in high seismic area, consider- able savings achieved for equipment support
M. B. Watson et al., Aerospace Corporation	5	5 to 10%; could be less than 5% if optimum use is made of the underground characteristics
Holmes & Narver, Inc.	47	\$382 per kilowatt total for underground plant in massive unfaulted rock

complete than a surface plant.⁴⁴ This estimate allows for 36 months of licensing requirements, including excavation for extensive site-investigation work. Further excavation to complete the cavities will take 18 months. All other plant construction work, which will start after the cavities have been essentially completed, is estimated to take 5½ years, which is 1 year longer than that for surface plant construction. Construction authorities are currently of the opinion that the actual plant construction need not take any more time than

surface construction, because the problems in logistics will be compensated by the less massive containment work and by the controlled environment in the caverns.

Therefore the lengthened schedule would be mostly in the licensing and site-exploration process, for which the financial outlay is considerable but nevertheless small compared with the total plant outlay.

CONCLUSION

Current technology is entirely adequate for placing a nuclear plant totally or partially underground. The need for certain engineering modifications has been identified, including ventilation, drainage, isolation, and high-pressure circulating-water systems. However, there is a real opportunity to utilize the characteristics of a rock cavern to design a less expensive containment and to better support structures and equipment. The overall safety is improved, particularly where severe accidents are involved. Although it is intuitive that an underground plant would be more expensive to build than a surface one, there are nevertheless many important factors that could offset the extra cost of rock excavation. The authors recommend that a serious evaluation program be initiated to delineate in detail the hardware and economic picture of underground siting. It is apparent from this study that the underground concept is a viable alternative to surface siting. Although the concept may be employed at any location, it may be particularly applicable in regions of high population density and/or high seismic activity.

Table 4 Excavation Cost Estimate

Item	Volume,* yd ³	Unit cost,† \$	Excavation cost, \$
Access tunnel	155,000	27.50	4,263,000
Vertical conduits and shafts	10,000	75.00 (50.00)	750,000
Horizontal conduits	15,000	(35.00 22.00)	525,000
Reactor containment	130,000	35.00	4,550,000
Balance of plant			
Heading	92,000	30.00 (15.00)	2,760,000
Benching	248,000	20.00 (10.80)	4,960,000
Other	120,000	25.00	3,000,000
Total	770,000		20,808,000

*Excavation volume is based on requirements for placing a 1000-MW(e) PWR underground.

†Unit costs in parentheses are from the Churchill Falls experience (1971).

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5.4 Comments Re the Article (in Vol. 15, No. 5) on Underground Nuclear Siting

[*Nucl. Safety*, 16(4): 434-435 (July-August 1975)]

In a letter to the editor, Mr. J. P. Buclin of Energie de l'Ouest-Suisse, Lausanne, Switzerland, listed 14 reasons why he is not convinced of the underground-siting advantages as set out in the article by Crowley, Doan, and McCreath on pages 519 to 534 in Vol. 15, No. 5, of *Nuclear Safety*. Mr. Buclin was the deputy head of plant engineering at Halden (1959-1962) and was subsequently the plant superintendent at Lucens (1964-1972). His 14 reasons are listed below, and they are followed by some comments by the authors of the original article and by the editor of this journal.

Comments by J. C. Buclin

1. None of the existing underground excavations, even those of considerable volume, have a span comparable to that needed for a 1000-MW(e) containment enclosure. Experience might show that the current technology is not adequate to meet this requirement in most types of rocks.

2. The saving of surface land is unlikely when one considers that the necessary surface construction area is of comparable size with the constructed area of a reactor located on the surface. Given current environmental concerns, the deposition of excavated material might even cause the greater problem with respect to surface-land use and environmental impact.

3. The use of a gastight liner around the nuclear cavities is mandatory, to fulfill the requirements for the containment of radioactivity releases. If the plant is built above groundwater (or rock water) level, as experience would strongly recommend, all ducts and penetrations must also be gastight, a condition which is more difficult to achieve underground. If a plant is built below the natural water level, which should be avoided in view of past experience, the liner must be protected from contact with groundwater, in which case the use of the rock as supporting medium around the cavity is lost and with it a main safety advantage of underground siting.

4. Poor accessibility, whether for inspection, maintenance, or repair, works to the disadvantage of underground siting and would tend toward increasing the frequency of small incidents at such plants. Furthermore, safety would not be improved for emergency conditions, i.e., a loss-of-coolant accident (LOCA) could be handled as well by a surface plant. However, safety could be enhanced for class 9 accidents, but in this situation the supporting action by the rock would be desired, and contact with groundwater should be avoided.

5. Underground siting might appear attractive for compact, potentially dangerous undertakings (not only nuclear!). However, it is not apparent that an acceptable emergency flooding system could be provided.

6. The use of rock saturation techniques to enhance the containment capability was originally anticipated at Lucens. It was not used during the course of the experienced LOCA, and would probably not have been used even if the LOCA had occurred at a higher burnup of the fuel. This is because uncontrolled spreading of fission products, by containment leakage into the groundwater, should be avoided as long as the emergency safety features perform satisfactorily. Use of rock saturation will represent an ultimate safety feature.

7. Drainage of rock water around the cavities and containment is not working well at many underground facilities, nor was it working at Lucens. As a result, one experiences corrosion of the liner, leakages, and corrosion of structures inside the containment and the cavities.

8. Earthquake design has improved much in recent years, and most sites which were questionable before (at least in Europe) seem to be accepted now using current technology.

9. The handling of heavy equipment at underground sites, particularly the removal of irradiated and contaminated equipment from underground facilities, appears to be one of the major disadvantages. Free access from all sides and by overhead cranes is a valuable capability for surface plants during plant construction as well as during decommissioning operations.

10. The use of high-pressure condensers, as required by most underground sites, adds to the costs and represents a continuing potential for failure of major importance to power plants designed for a service life of 40 years.

11. The acceptance factor of underground siting does not appear to be of great importance, since surface siting adequately close to load centers even for district heating purposes is permissible. An underground plant would require an elevated ventilation release point, even in the absence of any other surface building. This would compromise the acceptance factor.

12. Despite the experiences from the Plowshare program, the achievement of acceptable leaktightness following a class 9 accident in an underground plant appears to be very doubtful because of multiple accesses and penetrations. Sealing by flooding is probably the less attractive method because of problems brought along with the water and because of not solving the main problem brought about by leaks between the tunnel outside wall and the rock even if no other difficulties are encountered.

13. The experience at Lucens well illustrates the problems encountered to avoid water contamination: all contaminated parts have to be removed or decontaminated, so that the leakages are actually kept at drinking-water tolerance and are pumped away without any treatment. However, it must be pumped as long as any use is foreseen for these cavities.

14. Underground siting does not add much to immunity from some surface phenomena, except maybe from air crashes of very low probability. Ultimate heat sink and on-site power supply both cause difficult problems. The problem of providing protection of the normal heat sink and of the outgoing power lines is similar to that for surface plants.

Comments by the Authors (per J. H. Crowley)

We are grateful to Mr. J. P. Buchin for having shared his experience and views on the subject of underground nuclear plant siting. Our overall impression is that he voiced the same kinds of technical, safety and economic concerns which we have expressed either directly or subtly in our article. What follows are only some further explications of Mr. Buchin's remarks.

The subject of large cavity spans and geotechnical soundness as regards to seismic design for underground caverns has been described in our article. We are less concerned with the commonly expressed reservation in span size than with the earthquake qualification for a complex rock cavern dedicated to containing nuclear facilities. In practice, span sizes for hydroelectric caverns have not been very large because the equipment can be technically and economically housed in caverns with spans smaller than 100 feet. Recent development, as we pointed out in our article, indicates two facts: (1) span size for underground pumped storage reservoirs can be engineered to over 100 feet, and (2) nuclear primary systems can be modified such that very large cavern spans are not necessarily a prerequisite.

Mr. Buchin rightly pointed out potential problems with groundwater such as drainage, corrosion, water contamination, and rock saturation techniques. We believe that large nuclear facilities can learn a lot from the experience of the first few small underground plants in Europe, but that the technology exists to avoid repeating their problems without incurring forbidden costs.

The potential safety features of the underground nuclear plant are perhaps one of the dominant considerations. Based on our licensing experience and analytical analysis, we are of the opinion that the underground plant possesses an extra level of safety in the case of a Class 9 accident. In such an event, the underground plant is capable of delaying the release of fission products to the environment and thus enabling emergency measures and evacuation to take place. Mr. Bucin mentioned that nuclear plants have already been sited close to load centers in Europe, but this is not the case in the United States. The extra level of safety of underground nuclear plants should be of great significance for a site near a populous zone.

In conclusion, we again agree with Mr. Bucin's views and reiterate our position that there are advantages and disadvantages with the siting of a nuclear plant underground, and the subject should receive greater attention in siting considerations, research and development. From the economic viewpoint, there are many potential tradeoffs that will not be clear until a serious engineering effort will have taken place. The utilization of the depth dimension of our limited land area should be another incentive for serious consideration in view of the need for thousands of new power producing facilities by the close of the century and beyond.

Comments by the Editor

The editor wishes to express his appreciation to Mr. Bucin for his identification of various concerns which within his experience mitigates against underground siting, as well as to the authors of the original article, who have responded to Mr. Bucin's comments. In this, as in many matters related to nuclear safety, one relies heavily upon the subjective judgment of experts, but it behooves them to consider well all relevant experience. — *Wm. B. Cottrell*

5.5 Novel Siting Solutions for Nuclear Power Plants

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[Editor's Note: The College of Engineering and the Engineering Extension of the University of California at Berkeley sponsored a 2-day conference on Novel Siting Solutions for Nuclear Power Plants, held Nov. 8-9, 1973, in San Francisco, Calif. At the invitation of *Nuclear Safety*, the authors have prepared this summary of the lectures delivered and the discussions that took place. However, the authors go beyond the proceedings of the conference and attempt to evaluate the state of the art in novel siting.]

Abstract: This article is a review of the conference on Novel Siting Solutions for Nuclear Power Plants, held in San Francisco, Calif., Nov. 8-9, 1973. The state of the art of major technologies for offshore and underground siting is reviewed both in terms of decoupling the licensing of reactors from site-related environmental factors and engineering, financial, and regulatory uncertainties. Problems of comprehensive site evaluation and technology assessment associated with these innovative concepts are also discussed. The conference participants agreed that major new technologies for extending siting options are available. Offshore breakwater-protected power plants seem to enjoy almost universal acceptance among architect-engineer and utility planners, but opinions were strongly divided on the safety and economics of other novel concepts.

The nuclear utility industry is at a turning point in power-plant siting. Increased awareness and concern over environmental and safety factors and the industry's goal of siting hundreds of nuclear power plants in the next 30 years have forced planners to consider and occasionally embrace novel and unproved concepts. The industrial commitment is impressive. Manufacturers are accepting multiple firm-bid orders for floating nuclear units to be anchored offshore even before the construction-assembly plant is built. A parallel and interacting effort is under way to standardize plant design in order to expedite regulatory approval of additional units and to decouple the licensing of the reactor from specific site-related environmental considerations. However, inertia associated with conventional siting practices is well established in the utility industry, and novel siting will have to overcome the misgivings and uncertainties through demonstration of technical and environmental advantages and economic feasibility. Some major shifts in federal regulations will also be necessary. The

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lecturers, panelists, and their respective topics are listed in Table I.

NOVEL SITING OPTIONS

A number of novel siting concepts, all of which employ well-established reactor types, have been proposed in the last few years. The objectives of the concepts are to better isolate the reactor from the surrounding environment and to relieve environmental constraints while simultaneously conquering new siting space. For example, underground nuclear power plants, which presumably provide better radiological protection in the case of a major accident, could be located closer to urban areas and still provide safety assurance equivalent to that of more distant above-ground plants. Ocean-sited floating plants reduce the problems of waste-heat rejection, are seismically isolated to a certain extent, and extend the available siting space by including the ribbon of ocean near the coasts.

Table I Conference on Novel Siting Solutions for Nuclear Power Plants*

Topic	Speaker
Introduction	L. M. Grossman (University of California, Berkeley)
Site Evaluation Decision Methodology	B. J. Garrick (Holmes & Narver, Inc.)
Nuclear Power Plant Siting Concepts for California	Harold F. Paris (Holmes & Narver, Inc.)
Offshore Shallow-Water Floating Power Plants	R. C. Nichols (Offshore Power Systems)
The Atlantic Generating Station	Musaffir Kassarjian (Public Service Electric & Gas Company)
UCLA Study of California Floating Nuclear Power Plants	S. F. Palgr (University of California, Los Angeles) and F. J. Weller (Dames and Moore)
Panel Discussion: Offshore Power Plants	L. M. Grossman (moderator), B. J. Garrick, M. Kassarjian, R. C. Nichols, S. F. Palgr, E. J. Weller, and R. L. Wigel (University of California, Berkeley)
Geological Engineering Considerations in Underground Siting	T. L. Brekke (University of California, Berkeley)
Layout Considerations in Underground Siting	D. R. McCrath (AEC American, Inc.)
Engineering and Safety Considerations in Underground Siting	Noel Gonzaga (United Engineers and Constructors, Inc.)
Studies of Underground Siting of Nuclear Power Stations in Norway	Einar Jansen (Institutt for Atomenergi, Norway)
Panel Discussion: Underground Siting of Nuclear Power Plants	T. L. Brekke (moderator), Noel Gonzaga, Einar Jansen, D. R. McCrath, and Ray Blum (Pacific Gas & Electric Co.) Loren G. Hinkleman and R. P. Schmitz (Bechtel Corp.)

*Organizers: T. L. Brekke, L. M. Grossman, and G. Yadigaroglu.

A number of the proposed schemes are offshore options,¹⁻²⁵ which can be further subdivided into shallow-water and deep-water concepts.² Shallow-water options include natural or artificial islands,^{3,4} plants built on the seabed,⁵⁻⁷ and floating plants protected by breakwater.⁸⁻¹⁰ Both prestressed-concrete barges or steel-hull vessels and tuned spheres¹² designed to minimize motion have been proposed for deep-water sites.^{13,14,20,25} Both seabed-sited (lowered to ocean floor by compartment flooding) and under-the-ocean-floor options²⁴ have been considered. The latter would be constructed deep in the seabed from tunnels originating on the shore.

Underground siting options vary according to the ability of the geological strata to support themselves. Construction in excavated caverns or under hills, as well as burial after outdoor construction, has been studied.²⁴⁻⁴⁰ It is evident that a large number of concept and site combinations¹⁶ are possible (e.g., plants floating in a lake). As an extreme combination of floating and underground concepts, a nuclear station floating in water at the bottom of an underground chamber has also been considered.¹⁷

Siting schemes such as outer-space (satellite) nuclear power plants, which are feasible with present or near-future technology, have also been proposed.⁴¹ These and dual-purpose plants, such as power/desalination plants,^{3,4} and nuclear complexes combining power generation with adjacent immediate use, such as the nuclear energy centers proposed by the Oak Ridge National Laboratory,^{42,43} are not considered here.

Discussants at the conference quickly narrowed the range of novel options to those most technically and economically acceptable. Offshore floating and underground stations were placed in the perspective of imminent technologies. Underground nuclear stations and other massive caverns already exist throughout the world, and the construction of floating systems is in the late planning stage, with the first unit scheduled for commercial operation in 1980 and the second in 1981. Offshore and underground siting is discussed later in detail.

SITE-SELECTION METHODS

The selection of a particular power-plant site depends on interacting factors, including the choice of energy source and technology, the regulatory structure governing price and availability of various fuels, environmental policy and any uncertainty associated with it, and physical, engineering, ecological, and social aspects. Once a particular energy technology is chosen,

siting considerations focus on engineering feasibility and costs, public acceptance (including safety and aesthetic factors), potential startup delays, ease of regulatory compliance, and environmental impact. Location factors related to transmission facilities and load centers, access to transportation systems, cooling-water availability, geophysical terrain limitations, meteorology, and hydrology are the most critical of the physical factors.

The demography and ecology of the site strongly influence the environmental impact of the power plant. Utilities, reactor vendors, architect-engineering firms, and regulatory agencies have responded to recent environmental-impact reporting requirements. Early difficulties and delays have fostered a site-selection framework that is becoming increasingly standardized. Such a method has recently been applied to site selection in California.^{24,25} Similar methods are proposed in Refs. 44 to 49. An important part of the conference dealt with methods for incorporating new siting options in the overall nuclear energy system.

An explicit common basis for comparison of siting alternatives may provide the rationale for siting choice. In the systems approach suggested by Holmes & Narver,^{24,25} evaluation criteria were divided into the following four categories:

Evaluation category	Weighting factor
Cost factors	0.287
Environmental impact	0.301
Feasibility	0.272
External risks to facility	0.140
	1.000

In the absence of more objective information, the weighting factors were based on quantification of opinions held by a number of individuals involved in site selection, including spokespersons from industry, conservation groups, and utilities. The weighting used here implicitly includes a portion of the inherent nuclear risk of the facilities in the cost factor by requiring that all plants incorporate sufficient engineered safeguards to be licensable by the U. S. Atomic Energy Commission (AEC). Fuel transportation risks were not included in their analysis. Each siting concept is evaluated on a variety of bases within each category. For example, the procedure for estimating costs involves simply ranking the concepts according to estimated bus bar costs and converting those to relative numbers. If aboveground coastal plants produce power at 9 mills/kWh (the lowest cost option) and offshore

floating plants produce at 12 mills/kWh, the coastal rating would be set equal to 1 and the offshore rating calculated as 0.75 (9/12). These relative values within each siting concept are then multiplied by the weight of the appropriate evaluation category and summed. The overall ratings of the various siting concepts are then compared, and the concept with the highest numerical rating is the preferred one. A number of graphical display techniques are also used to contrast acceptability of all factors individually as well as in a weighted sense.

Since the costs are converted to relative numbers before weighting, the choice of options considered for analysis can change the acceptability ordering, thereby creating the dangerous illusion that the assigned weights directly incorporate public preference. The procedure has additional operational problems; for example, Holmes & Narver suggested that when "miles of potential sites" were used as a measure of the availability of sites, it was necessary to "deemphasize" the availability of siting concepts with many alternate locations. This subjective action effectively introduces a secondary weighting factor to the already questionable procedure.⁸ When this procedure was applied to the California case study, the analysis indicated that inland siting concepts were the most favorable; other high-ranking concepts included coastal-zone plants, plants aboveground in a hillside setting, and platform-mounted plants in lagoons. It is interesting to note that many concepts ranked well.

Although the weighting scheme lacks objectivity and an accepted theoretical basis and does not consistently value options according to the initial weighting factors, the conference attendees agreed that the scheme is accepted by the AEC for identification of general alternatives in long-range planning. Significant additional conceptual problems remain concerning the separation of co-related categories, such as cost factors, feasibility, and safety, and the choice of categories and weights. An expanded systems procedure might treat cost factors and feasibility as a continuum with inexpensive, readily available, and proved technology on one extreme and speculative, high-cost, and undeveloped technology on the other. This framework would not only explicitly state technical and social assumptions but would also identify areas of fruitful research effort. For example, the framework might address the questions as to what cost reduction or

⁸DiManno and Elkins²⁶ support an even more unusual weighting approach using varying regional weighting factors. Dehon et al.²⁷ recommend weighting factors that give equal importance to economic, environmental, and sociological considerations.

institutional change would make each option the optimal siting scheme or what aspect of a given technology would be most amenable to research and engineering advance.

The adoption of a novel siting concept and technology may increase the time period before plant operation but may also eliminate some delays that have become routine in conventional siting. Therefore it is important that utility planners correctly assess the costs of the two categories of delays. For example, the increased environmental reporting required for novel siting options could extend the preconstruction planning interval. If a project is on a tight time schedule, these delays push the operation date beyond the date that would maximize utility revenues and/or profits.* This potential delay could also be important from a social perspective. If this delay is more costly (in terms of profit and nonprofit corporate motives, such as the utility's desire to meet rising power demands) than the consequences of earlier project initiation, the utility should act accordingly.

OFFSHORE POWER PLANTS

Offshore nuclear power-plant concepts vary according to the coastal-water depth.² Shallow-water floating plants protected by surrounding breakwater⁸⁻¹⁰ are commercially offered for East and Gulf Coast locations, where broad expanses of shallow water within the 3-mile jurisdictional limit exist. Deep-water floating power plants have been considered for West Coast locations by a University of California, Los Angeles (UCLA)—industry study group.^{13,14}

Shallow-Water Floating Power Plants^{8-10,15}

This concept, pioneered by Offshore Power Systems (OPS),† emerged when a proved and licensable land-based nuclear power-plant design [pressurized-water reactor (PWR)] was applied to a floating plant.

The first two floating units have been purchased by the Public Service Electric & Gas Company (PSEG) of New Jersey to form the Atlantic generating station. An

artist's view of Atlantic 1 and 2 is shown on the cover of this issue of *Nuclear Safety*. Options or firm commitments by PSEG and other East Coast utilities to purchase additional units may indicate a new siting emphasis all along the East Coast. Offshore Power Systems is offering the nuclear units on a firm-price contract basis. The additional breakwater construction costs have been estimated at \$215 million for a two-unit station. The costs vary with the third power of the water depth. Single-unit stations are economically unattractive because of disproportionately high breakwater costs.

The specific details of the barge, plant design, and layout are in the final planning stages. Each PWR-generator unit will be mounted on a steel barge structure in an ice containment. Work on the first floating unit will proceed while the manufacturing facility is being completed at Blount Island, Jacksonville, Fla. The facility could eventually produce as many as four assemblies each year. Completed units will be towed by tugs to final river or ocean sites, where they will be moored by articulated arms within permanent breakwaters in water 40 to 70 ft deep.* Each barge unit is a complete integrated nuclear generating station and will be functionally hot tested at the production plant before site installation in pairs within a breakwater. The fuel will be loaded after the plant reaches the site.

At the conference, Nichols discussed the number of design alternatives that OPS considered during the early stages of design. A floating plant was preferred to a grounded plant because less extensive bottom preparation is necessary and siting flexibility is increased. Differential settlement problems are eliminated, and cathodic corrosion protection and seismic-response problems are alleviated. However, the floating concept introduces more stringent power-transmission problems and the safety-related questions regarding possible sinking. A steel hull was preferred to a concrete hull to take advantage of lighter weight (smaller displacement and draft for a given size) and easier fabrication, inspection, and repair. A square barge was preferred to a long rectangular one to increase stability, to reduce plant motion, to minimize hull stresses during towing, and to improve flexibility in plant layout. Finally the decision to surround the plant by a breakwater was made to simplify plant, hull, mooring, and electrical-connection design; to protect

*Utility profit may be calculated from the net discounted present value of the project to the utility under a variety of investment and revenue schedules and regulatory fair-rate-of-return provisions. The utility cost of delay is calculated by contrasting the possible scenarios. The cost of delay is less if the delay occurs during early planning stages than if it occurs after major investment begins.

†OPS is a joint venture between Tenneco and Westinghouse Electric Corporation.

*First-generation plants will be limited to depths that prevent complete submergence but provide at least 10 ft of bottom clearance.

the plant from large-ship impact; and to more closely approximate land conditions for plant operation.

The floating system may offer new benefits and options that are unavailable in land-based systems. First, the concept takes advantage of reduced manufacturing costs, improved quality control, and the shorter plant installation possible in a shipyard-type assembly-line facility. Second, the units⁵⁰ enjoy siting flexibility within a zone of coastal, federal, and state jurisdiction.* It is claimed that the wide latitude in potential locations facilitates the choice of sites^{10,51} that minimize environmental impact.^{10,19-21} The avoidance of delicate near-shore ecosystems is an important consideration.

The third advantage is the minimal impact on land and water uses and the ease of property acquisition without the high cost of purchase. The floating system uses 70 to 90 acres for a two-unit plant rather than a site commitment of approximately 500 acres. This advantage may be fictitious in light of the much larger land commitment necessary for power lines under any siting scheme. In addition, the land commitment for the offshore plant, shore-based support facilities, and the quarry acreage needed to supply the breakwater fill material may exceed that for conventionally sited plants. It is likely that the true land-related advantage is not a reduced land commitment but rather a minimization of problems of exclusion zones and competing land uses near the site.

The fourth advantage of floating plants is the complete adherence to the standardized design concept. The AEC policy statements issued May 1, 1972, and Mar. 5, 1973, indicate that standardized design plants may receive special expedited review in a two-stage permit process.^{52,53} After a plant concept has been approved and a manufacturing license has been issued, any particular power-plant project using the standardized design would only need to satisfy regulations focusing on the localized impact on the site environs. The amendment proposal, published in the *Federal Register*,⁵⁴ specifically described the OPS approach. Perhaps this amendment recognized that the floating concept was a major stimulus of the new regulations.

Plants sited in the ocean will face a much more hostile natural environment and must be protected against ocean storms and waves, hurricanes, tsunamis,

etc. Geological, meteorological, and ecological survey techniques necessary for offshore site investigations are also more complex and expensive than those required for onshore sites.¹⁰

Deep-Water Floating Power Plants

The UCLA industry study of deep-water floating power plants^{13,14} was conducted in the format of a case study of West Coast feasibility. In contrast to the Atlantic Coast, which has a broad and shallow continental shelf, the West Coast has a very narrow shelf immediately adjacent to deep waters. Areas of the California coast were initially categorized on the bases of ocean depth, ocean floor material and topography, proximity to land-based power stations, shipping routes, subsurface currents and temperatures, and storm conditions. The proposed plant was a barge-mounted unit similar to the OPS power plant. It would be surrounded by a 1.5-mile-diameter floating buoy-and-net protective barrier that would be anchored to massive drag weights on the ocean floor.

Safety and Engineering Considerations

The floating-plant concept is not without problems. Important safety and environmental engineering questions remain to be solved regarding response to seismic activity and storm waves, barge motions, ship collisions, and to the reliability of the off-site power supply via a submerged cable.^{10,51,55-58} In particular, the flexible power cables necessary to connect the plant to the undersea cable must be further developed. Some other relatively minor technological innovations are also necessary. Social-acceptance problems will certainly intensify if these plants present an increased risk and have a detrimental visual impact.

Seismicity remains the most critical site-selection problem, particularly for California and, to a lesser degree, elsewhere. A floating plant would provide a decoupling effect for horizontal movement, although a fraction of the horizontal forces will be transmitted through the mooring system. Vertical acceleration transmission would not necessarily decrease. Further research is necessary to determine the consequences of vertical accelerations on large floating structures. Some protection against vertical acceleration could be obtained through flotation of the barge on air springs or a spherical bottom design.¹⁷ The breakwater also must be capable of withstanding seismic forces. Similarly the influence of waves 50 to 60 ft high must be evaluated both during normal operation and while the barge is being towed to and from the site.

*Offshore site evaluations have neglected international water location (beyond 3 miles) on the grounds that legislation prevented a siting outside the coastal zone. This omission might be later reviewed by regulators and intervenors.

The barge motion must be subdued during normal operation and also during the critical refueling period. This is certainly a major engineering problem for open-ocean plants⁵⁶ and, to a lesser degree, for breakwater-sheltered plants. A maximum oscillation amplitude of 1.5° has been measured on models of the Atlantic generating station, which is designed for accelerations resulting from up to 3° single-amplitude motions at 13-sec periods. Potentially significant wave action may be generated within the breakwater during seismic activity from breakwater motion.¹⁵

Conference participants quickly pointed out that the floating concept adds an additional accident mode, ship collision,⁵⁷ and panel members agreed that the major danger of such collision is not the impact, which will be mitigated by the breakwater or other collision barriers, but rather the consequences of flammable or explosive cargo releases. Panelists also remarked that the probability of a ship penetrating the breakwater through the opening is not completely negligible and that additional safeguards are needed to guard the platforms against this possibility. Atlantic 1 and 2 will have a series of impact barriers installed near the breakwater opening to arrest any large ships that might penetrate these openings.

If no ship-collision barrier is provided, the probability of a stationary platform being struck by a ship of over 1000 tons is 10^{-3} to 10^{-4} per year.¹³ This probability is influenced by the proximity of the site to shipping lanes and by the volume of shipping traffic. The potential damage is increasing as ships become larger and incorporate strengthened hulls, such as the Arctic petroleum tanker ice-breaking bows. The possible future appearance of submarine tankers should also be considered. The most dangerous accident might be a collision of a petroleum or liquefied natural gas tanker at maximum velocity followed by fire. Incident radiative heat fluxes ranging between 10^4 and 10^5 Btu/(hr)(ft²) have been estimated¹³ for 3000 to 5000°F equivalent-blackbody-temperature fires covering 1 to 2 million ft² at distances ranging between 2000 and 6000 ft from the barge. A survey of light-petroleum-product fires shows, however, that, for more reasonable equivalent-blackbody fire temperatures in the range 1600 to 2000°F, the heat fluxes on objects in contact with the flames would not exceed 30,000 to 50,000 Btu/(hr)(ft²). These heat fluxes decrease rapidly with distance from the fire. Although such fires could not be extinguished, the barge wall temperatures could be reduced to acceptable levels by external cooling equipment. Atlantic 1 and 2 will rely on sealing booms to keep the flammable materials at a

distance and on fire-fighting equipment. Coastal-sited power plants may be similarly endangered by ship collisions and explosions or fires, but the probability is less than that for offshore platforms, the setback from breakwaters is greater, and the access to fire-fighting equipment and support is greater.

The OPS plants are to be built to the same safety standards as land-based plants. No essential differences in the engineered safety systems are planned, but the units must have additional vital systems to assure a safe shutdown if the plant sinks. This requirement has been imposed by the AEC in spite of the manufacturers' claims that no single mechanism or series of events could lead to sinking of the platform.

If ship collisions or other marine hazards do not jeopardize the nuclear station of an offshore plant, the probabilities of major radioactivity releases (Class 9 accidents) remain similar to those of land-based facilities. However, the consequences of such accidents are not equivalent. That is, core meltdown and foundation penetration on a land-based plant might result in extensive loss of life and related property and environmental damage. The same accident in a facility floating on an ocean or other body of water might have more serious consequences because the beneficial retention and filtering capability of the ground is lost. However, the immediate consequences of airborne radioactivity would be lessened since the water would retain some of the escaping gaseous or volatile fission products, notably the radioiodines. A 75% core meltdown and release of radioactivity in the Atlantic Ocean after a 1-year irradiation time would release 2×10^6 Ci of ⁹⁰Sr inventory. Approximately three hundredths of 1% of the total Atlantic Ocean water volume (10^5 km³) would be necessary to dilute this single isotope to the maximum-permissible-concentration level.⁵⁸ The ocean currents and turbulent diffusion characteristics would determine the possibility and expected interval necessary for this dilution.¹⁹ The public attitude toward consumption of potentially contaminated seafood may extend the economic loss beyond the value of actually contaminated marine populations.

Atmospheric diffusion characteristics are also generally less favorable over the ocean surface.⁵⁹ However, this factor is largely compensated for by the greater expected distance from inhabited areas on the coast. If manufacturers are required to reduce the probability of extreme events sufficiently to counterbalance any increased consequences and arrive at the same total risk as for land-based plants, the technology might be too expensive. Advocates of the floating

concept claim that quality control improves in shipyard manufacturing as compared with on-site construction. If this is true, the implications would have an impact not only on the acceptability of the floating systems but also on the licensing of conventional land-based nuclear plants whose quality control has been assumed to be very stringent.

Finally the panelists discussed potential psychological effects of long isolation periods on plant personnel, especially during inclement weather. However, experience from similar operations, such as ocean drilling platforms, indicates that these problems are not serious.

Visual Impact and Ocean Use

The prospect of a series of breakwater-surrounded power plants 3 miles from the shore may not receive universal public acceptance. The structures and use of the ocean might have to be considered more permanent in the environmental analysis for breakwater-surrounded plants than for land-based plants. The likelihood and feasibility of decommissioning breakwater-surrounded plants are not as well understood, and a major advantage of the concept may be the replaceability of the floating plants when economically desirable. The large site development costs of ocean locations would encourage perpetual use.¹⁵

In visual simulation using an aircraft carrier, a floating plant at 3 miles was found to be an imposing,

recognizable object from the sea-level perspective. The curvature of the earth will screen about 25 ft of the 180-ft-high plant, leaving about 35' of arc above the horizon. In West Coast sitings the intrusion would be highlighted in silhouette during sunset. As the altitude of the observer increases, the impact remains, but it is slightly reduced as the horizon appears above the structure. Aesthetic motives would favor camouflage to reduce the impact, but safety considerations dictate visibility to ships and aircraft.

UNDERGROUND POWER PLANTS

The concept and layout of underground power plants²⁶⁻³⁰ will vary according to the characteristics of the rock mass. Deep underground self-supporting caverns have been and can be constructed in massive rocks.²⁶⁻³² Poorly cemented or unconsolidated materials dictate construction of massive buildings capable of withstanding soil pressures and uplift forces at relatively shallow depths.³³⁻³⁵ Combinations of the two concepts with the reactor building just below the ground also exist.^{34,35}

In contrast to the offshore floating nuclear power plants, which are still in the design stage, several small underground plants already exist (Table 2).³² There is also accumulated experience in the excavation of rock caverns for underground hydroelectric, military, and other projects. Large underground caverns having spans

Table 2 Significant Features of Existing Underground Nuclear Power Plants or Reactors³²

Name and location	Size	Purpose	Configuration		Depth, ft	Reactor chamber dimensions, ft
			Turbine generator	Reactor		
Halden ³¹ Halden, Norway (BHW [*])	25 MW(t)	Experimental	None	Rock cavern	98	98 long 85 high 33 wide
Agesta ³³ Stockholm, Sweden (PHWR [†])	80 MW(t) 20 MW(e)	Heat and power	Aboveground at reactor grade level	Rock cavern	49	88 long 66 high 54 wide
Chooz ³³ Ardennes, France (PWR)	266 MW(e)	Power	Aboveground	Rock cavern		138 long 146 high 69 wide
Lucens ³⁴ Switzerland (pressure tube, D ₂ O moderated, CO ₂ cooled)	8.5 MW(e) 8.5 MW(e)	Experimental power	Rock cavern	Rock cavern	120	54 long 90 high 54 wide

*Boiling heavy-water reactor.

†Pressurized heavy-water reactor.

up to 95 ft (Sacramonedo, Italy), heights of up to 175 ft (Boundary, USA), and lengths of up to 1000 ft (Churchill Falls, Canada) have been successfully excavated.²⁰ [Editor's Note: *Nuclear Safety* 15(5) contained an article on "Underground Nuclear Plant Siting."]

Underground plants sited at shallow depths in unconsolidated materials obtained an overall ranking higher than the deep-rock-sited plants in the California siting study.^{24,25} However, this was mainly due to the fact that 10 times more sites of unconsolidated or poorly cemented materials exist in California. The California study also assumed equal nuclear risk for all plants through the use of appropriate engineered safeguards. It seems that power plants built in rock caverns would offer a higher degree of safety and environmental protection at a higher construction cost. A detailed cost-benefit analysis would be needed to determine whether this trade-off is socially beneficial and whether the extra costs could be utilized more effectively in improving the safety of conventionally sited power plants. Unfortunately, at the present time, no clear-cut indication of the worth of environmental and safety considerations is available.

Although floating plants were considered by conference participants with optimism and a near-unanimous support, some of the opinions expressed on underground plants were diametrically opposed. The proponents of underground plants claimed that such plants provide economically and technically feasible alternatives and that, although unsolved engineering problems exist, technological solutions are not insurmountable. They pointed out that urban siting of underground plants, although a future possibility, might not be feasible with the present licensing criteria in the United States. However, representatives from other architect-engineering firms firmly contended that present efforts to build underground units in this country would not produce benefits in excess of costs.

The major advantages of underground nuclear power plants would be related to safety and aesthetics. These advantages could, in turn, yield additional economic benefit: through the possibility of urban or near-urban siting. Economic advantages include lower land-acquisition costs due to smaller land commitment. Reduced impact of the weather conditions during construction might be a significant advantage in some climates. Aesthetic advantages are possible through the absence of large industrial structures in rural settings.

Underground plants could provide equal and possibly better containment than aboveground plants.

They are also much better protected against such natural surface phenomena as storms and tornadoes and such manmade threats as airplane crashes, acts of war, and, to some extent, sabotage. Biological shielding and protection of the primary system from turbine missiles would be enhanced by thick cavern walls. Finally some studies indicate that better seismic protection of underground plants is possible.³²

Disadvantages of cavern-sited underground plants are mainly cost related. Costs extending well beyond the direct excavation and rock reinforcement costs (rock bolting, shotcrete, etc.) must be considered. Longer construction schedules and possibly increased site-survey efforts will also result in higher costs. The tight sequence of construction scheduling and the coordination of the underground movement of large quantities of materials and equipment is a serious problem. There is a need for numerous ventilation ducts, passenger and freight elevators, sophisticated isolation and sealing systems, and a high-pressure condenser design. These items and the generally increased piping, ducting, and cabling lengths, and possibly increased labor costs, will add to the capital costs.²⁰ Plant maintenance will be more difficult underground. The risk of flooding the underground cavities associated with cooling-water-line ruptures is one new significant risk introduced by undergrounding. This would not be the case for hillside plants. There might also be increases in loss-of-ventilation risks, which would affect control-room habitability following an accident.

Conference participants agreed that large underground nuclear power plants could be constructed within the present state of the art. However, some extrapolation of maximum cavern spans and research related to isolation and sealing systems for access shafts and tunnels is needed. With the present licensing regulations, a practical limitation to underground siting might stem from the impossibility to preclude discovery of nondatable geological faults during excavation.

As reflected in the breakdown of the lectures, underground siting problems can be classified under the general headings of geological engineering, plant layout, and safety considerations. Some particular aspects of underground siting are discussed below in greater detail.

Underground Layout

In contrast to the regimented standardized plant layout of a large-mounted plant, underground siting in

rock requires significant layout tailoring. The layout must fit within the site-dependent geological regime; enable efficient excavation, construction, and erection; and satisfy all operational and safety requirements. In addition to the layout problems common to all large excavations, the anticipated high cost of separating individual cavities resulting from long piping runs and the associated piping-failure probability must be considered. However, underground siting does offer the potential for three-dimensional layout (Fig. 1), which

somewhat similar concept has also been proposed for a coastal urban site.³³ In the LLL study the reactor containment is placed at a depth of several hundred feet and covered with backfill having controlled permeability, and the turbine building is located at a small depth.

The additional pumping power needed for condenser cooling varies linearly with depth and is a decreasing function of the allowable water heating range, ΔT . It is estimated to be approximately 2% for a

Fig. 1 Layout of an underground 1000-MW(e) nuclear power plant constructed in rock. (Courtesy of Acres American, Inc.)

may partially offset the increased costs of excavation, assembly, etc. Economic considerations suggest long, high tunnels of constant cross section, but other, more complicated, cavity shapes might be better suited to fit the in-site geological and stress conditions.²⁸ Rock cover of the order of twice the maximum cavity span is required to achieve the effective development of self-supporting rock arches. However, excessive plant depth will increase access-shaft lengths and costs and will result in even higher pressure design of cooling systems. Approximately 300 ft of cover is probably reasonable, both in terms of technical adequacy and economics.

A recent Lawrence Livermore Laboratory (LLL) study³⁴ proposes open-pit excavation to allow construction of an underground reactor containment structure in almost any geological medium (Fig. 2). A

depth of 300 ft and a ΔT of 15°F. The pressure losses resulting from vertical separation of the reactor and turbine buildings are estimated to be less than 1% of the reactor pressure, whereas heat losses are less than 0.04% of the gross electrical output under worst-case assumptions.³⁴

Containment

First-generation underground plants have been enclosed in a containment or constructed in lined cavities where the lining replaces the conventional pressure-containment envelope. Therefore these plants have not relied on any surrounding rock masses for pressure accommodation. However, the ground will play the role of a secondary containment and minimize leaks to the surface. In all cases it is desirable to build the reactor containment at a sufficient depth to ensure

Fig. 2 Buried underground plant proposed by LLL.³⁴ (Courtesy of Lawrence Livermore Laboratory.)

that groundwater pressures will always be in excess of any possible accidental overpressure within the reactor chamber.

Advanced underground containment designs attempt to place the reactor in an unlined rock cavity and would rely on the retention capability of the rock mass to limit leaks of any radioactivity to the surface.³⁰ Measurements of granite permeability are reported in Ref. 60. At the conference, Jamne presented the results of recent (1971) field experiments on rock permeability performed at a potential underground reactor site in Norway. The rock permeability increased near the surface, and leak rates calculated using the measured rock permeabilities typical of depths below 80 ft were in the vicinity of 0.2 to 0.3% per day per atmosphere of overpressure for a rock cover at 250 ft. However, this result was obtained by assuming that the rock in the vertical projection of the reactor wall was completely dry. A modest groundwater pressure would have effectively prevented the out-leakage. This observation led to the suggestion by Lindbo (as reported in Ref. 32) of a three-dimensional water curtain surrounding the reactor cavity which would provide an artificial groundwater pressure in the absence of a naturally high groundwater table. A drainage system surrounding the reactor cavity and kept at subatmospheric pressure would preclude contamination of the adjacent groundwater.

The presence of fissures in the rock mass might be more limiting than the penetrability characteristics of the rocks discussed above. If rock fissures cannot be ruled out, grouting and steel lining of the cavity are necessary. A conservative accident analysis should assume leakage through fissures and ground-level dis-

persions. The Norwegian work reported at the conference also indicated that filtration and delay effects due to the passage of any releases through fissures in the rock overburden might be of considerable interest for a rock containment.

The fission-product retention and filtering capability of ground rock is considered in some detail in Ref. 31. Rock masses having even modest porosity provide an enormous volume for expansion of steam escaping from a loss-of-coolant accident (LOCA). Similarly the extremely large surface areas presented by porous rock (5,000 to 30,000 ft²/ft³) will be beneficial for condensing steam and adsorbing fission products. Under those conditions, only noble gases would leak out, first driven by the containment overpressure and later, after decay of the containment pressure, driven by diffusion only. Leakage properties of rock are also reported in Ref. 61.

The LLL study of a "buried" underground plant³⁴ includes an investigation of the releases through the backfill material following a LOCA. In this case a 10-ft layer of low-permeability backfill prevented leakage to the atmosphere even under the worst conditions. The gas front could propagate only up to distances of the order of 10 ft before the containment pressure decayed to atmospheric levels.³⁴

On the negative side, panelists noted that little design credit would be allowed by current AEC regulations for diffusion of releases through the ground and that plants having "zero" routine releases are possible aboveground. Furthermore, any beneficial building wake effects in diffusion calculations are lost.

Contamination of groundwater, not considered a serious problem, could be prevented by the design of a

drainage-sump system and the incorporation of emergency purification and drainage systems capable of keeping the water level below certain limits for an indefinite period following a LOCA.

Containment Isolation

Any beneficial effects of release retention in the ground would be lost if a reliable and effective containment isolation is not achieved in the case of a major accident. In fact, it was noted that conventional containment buildings leak through their penetrations and that similar numerous penetrations for cables, ventilation ducts, access shafts, etc., are inevitable in underground plants. It is surprising that most of the underground containment studies have concentrated on diffusion through the ground and have treated the containment-isolation problems rather superficially.

In the two-step containment-isolation procedure proposed in Ref. 28, different accesses, ducts, and roadways are first isolated by closing valves and then are flooded from above to form a water seal. In the case of an accident followed by malfunction of the containment-isolation systems, the leaks into the access shafts and tunnels could also be collected, filtered, and temporarily stored in the large shaft and tunnel volumes.³⁰ Leakages in the rock adjacent to the air locks could effectively be reduced by injecting concrete and clay and by applying a curtain of pressurized air in the rock in this region.³⁰ Thus it seems possible to obtain an effective secondary containment at a reasonable cost.

The LLL buried plant relies on fast-closure systems developed for nuclear detonation tests for isolation. It is claimed that such systems are reliable and can withstand extreme loadings.³⁴ It appears that part of the uncertainty of the effectiveness of containment isolation could be resolved through a detailed probabilistic analysis of the reliability of the isolation systems.

Seismic Considerations

An underground plant is not immune to seismic forces. One argument in favor of underground siting of nuclear power plants has been that an underground plant is inherently safer from earthquakes than a surface plant. Studies summarized and analyzed by Brekke and Glass³² indicate that the amplitude of seismic motions is less at depth than at the surface. However, this relation appears to be frequency dependent, and studies or measurements of the frequency characteristics of strong earthquake motion at depth

are lacking. Also, the frequencies for which motion is reduced at depth might be above the natural frequencies of major equipment.

In his presentation, McCreath noted that underground siting raises the interesting possibility of modifying certain aspects of the internal plant layout to enable bracing of equipment and structures back to the bedrock over their full height to avoid the effects of in-plant amplification of seismic motion. This factor may reduce the design accelerations for various items of equipment and reduce their cost.

Underground sites would have to comply with current regulations regarding active faulting.* In this respect, there is no siting advantage underground. On the contrary, as stated earlier, problems of dating any faults discovered during excavation may make siting infeasible, although faults may be avoided or the plant layout may be modified to bypass faults through noncritical areas. Any seismic problems encountered during excavation would certainly result in costly planning, licensing, and construction delays.

Other Safety Considerations

Underground siting of a nuclear power plant eliminates some critical accident paths, reduces the probability of others, and creates or increases the probability of still others. For example, it might be expected that underground equipment would be sufficiently isolated from the influence of natural meteorological phenomena and airborne objects, thus eliminating the possibility of direct equipment damage from surface events. However, underground siting of a plant may increase the length of reactor coolant piping, thus increasing the probability of line breaks. The probability of turbine-blade failure is independent of plant location, but the effects could be mitigated by separating the rock mass from the critical components. New accident potential or consequences may be created or increased by underground cavern or coolant-line failures, leading to flooding of critical components, or by the reduced physical accessibility of plant components. However, this same inaccessibility is a recognized advantage in controlling access of saboteurs. These advantages and disadvantages must be assessed.

Further unanswered safety questions include underground control-room isolation, spent-fuel-cask handling, ventilation-systems sealing, access after acci-

*Defined as a fault zone that has exhibited movements once in the last 35,000 years or more than once in the last 500,000 years.

dents, and reliability of piping to vital emergency heat sinks. The heat buildup in the rather well-insulated underground containment following a LOCA should also be considered.

Underground containment intuitively appears to offer at least partial solutions toward the mitigation of extremely serious (Class 9) accidents, such as uncontrolled LOCAs or pressure-vessel failure. In addition to the fact that releases from molten cores would be probably better confined underground, underground containment offers a better possibility of gravity-fed emergency core-cooling systems. The operation of such systems will be limited, however, by flooding of the containment and will require recirculation cooling for some time after the accident. The possibility of flooding the entire underground complex has been examined²⁸ but should probably be avoided since the air trapped in the containment cannot be released. Thus complete flooding will raise the internal containment pressure and increase the outward leakage of the airborne activity as well as contaminate the groundwater. Finally, Swedish studies⁴² point to the capability of underground rock containments to withstand the missile forces and steam shock waves created by a vessel failure, as well as the impacts created by sabotage explosions. The report states that the cavity walls will move very little because of the large masses supporting them. Thus tight membranes applied to the cavity walls could maintain their integrity after a violent accident.⁴⁴ However, the question of the integrity of the membranes will depend on the method of attachment to the cavity and is open to challenge.

Economics

Various studies have estimated the differential costs of siting a conventional nuclear power plant underground.^{31, 33-35} These estimates range from a few to 50% of the cost of the conventional plant.

To establish the economic desirability of underground siting requires evidence that no increased safety margin can be achieved at lower cost through other siting or engineering options. For example, if an urban underground site were evaluated, the cost of more remote siting should be contrasted with additional costs of underground excavation, construction, and operation. These were the areas in which the undergrounding concept met with stiff criticism. One panelist estimated that as much as \$10 to \$20 million would be necessary for a suitable exploratory excavation. Others expressed concern that underground construction and assembly would entrain many more

delays* and could cost as much as 50% more than that for a surface plant, partly due to a 1-year schedule penalty. Proponents of underground plants estimated that the civil-engineering costs would be 25% higher for an underground plant on the basis of experience with large underground hydroelectric projects. The audience quickly conjectured that no company would guarantee firm unit costs, as is now done by OPS for the floating units. The cost of breakwater is less uncertain than the costs of exploration and adoptive underground excavation.

Another potentially beneficial economic uncertainty of the underground concept is whether the need for environmental studies will be reduced or limited to a shorter time interval.

CONCLUSION

Only two novel siting concepts have been explored comprehensively at the present time. A number of floating nuclear units will be under construction in the near future, although the entire concept of large floating power systems has obviously never been tested before. Conversely, there are no plans for building large underground nuclear power plants in the United States, although their feasibility has been successfully demonstrated abroad, and they will probably enjoy good psychological acceptance.

All the studies mentioned here have considered siting of well-established reactor types, such as the light-water reactors. Although there is probably no incentive in siting more advanced concepts, such as breeder reactors, offshore, the underground siting of liquid-metal-cooled fast breeder reactors might be advantageous.

A variety of studies are needed to investigate particular aspects of novel siting. It appears that it industry's obligation to provide alternatives to conventional siting, but it also seems that some aspects of the present regulatory framework and the inertia of the utility industry might inhibit desirable changes.

*For example, an underground site will require a rigid sequential assembly of major structures and equipment. A supply delay or sequence error might require the curtailment of the entire project and its associated costs. Since many states do not allow the addition of nonoperational capital in the rate base, this additional cost might be substantial in terms of utility profits. These problems, however, might not be unique to underground siting. A centralized power-plant manufacturing facility may also suffer from component shortages, scheduling problems, prolonged labor disputes, natural disasters, etc.

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5.6 Nuclear Energy Centers: A Prime Element in Reactor Siting

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Abstract: Meeting the nation's future energy requirements portends a major role for nuclear power. This could result in hundreds of relatively small, dispersed nuclear sites throughout the country by the year 2000 if present siting trends were to continue. The siting of several nuclear reactors and their associated facilities in a small number of concentrated nuclear energy centers is an alternative having many advantages. This article describes the background of the idea and discusses some of the advantages and problems in clustering nuclear reactors into nuclear energy centers. It concludes that there are compelling arguments for the national deployment of nuclear energy, particularly breeder reactors, into such centers but notes that there are many unresolved issues on which much work must be done before final answers are available as to the size, composition, and extent to which the energy-center concept can be applied.

Although nuclear fission at present provides only a small part of the U. S. energy requirements, almost every scenario on the nation's future energy requirements projects a major role for nuclear energy. The nation is well along toward commitment to a nuclear power economy, possibly for a very long time (burner reactors for the near term and breeder reactors for the long term). Thus it is essential that sound and careful thought be devoted to assuring that nuclear technology is developed in a way that will maximize the benefits with acceptable environmental, social, and economic risks. Much has been done to this end, and no other field of technical endeavor has received as much attention and careful planning. However, there are still issues to be resolved and much to be done to improve the role of nuclear energy. The purpose of this article is to treat one such issue: the siting of nuclear reactors.

The current practice for the siting of nuclear reactors has resulted in a number of dispersed sites (approximately 120) being developed throughout the country. The projected nuclear power capacity of perhaps 10⁶ MW(e) or more by the year 2000 will require about 900 reactors in operation and perhaps

another 500 or so under construction. Thus we can postulate that there may be about 1400 nuclear reactors operating or under construction by that time.

The present tendency is to disperse reactors, although as many as four to six per site are being planned for a limited number of sites. Therefore the average number of reactors per site will be small for a prolonged period unless the tendency to clustering accelerates significantly. An assumed average of three reactors per site would result in about 470 sites (almost four times the present number) by the year 2000, with the number increasing rapidly thereafter. This raises the question as to whether the proliferation of nuclear reactors over hundreds of sites throughout the country within the next 40 to 50 years is an acceptable long-range practice and suggests an important and perhaps essential step to improving the posture of nuclear energy through the establishment of nuclear energy centers. An artist's conception of a nuclear energy center on a shore site with once-through cooling is shown in Fig. 1, and a similar sketch of an inland site with wet cooling towers is shown in Fig. 2.

The maintenance of a healthy and growing nuclear power industry is an accepted means for meeting the country's near-term energy goals as well as its long-range energy requirements to the year 2000 and beyond. The nuclear electric capacity planned and under way is expected to be about 30% of the country's installed electric capacity by 1985. It is projected to increase¹ to more than 50% by the year 2000 and even more rapidly thereafter. The recent deferrals by utilities of nuclear plants planned for construction generate some uncertainty on nuclear-power-growth projections. However, unless these deferrals are indicative of a major change in electric-power-growth trends, they will merely shift the time schedule by a few years without altering the basic problem. If this increased power growth should be

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Fig. 1 Shore site nuclear energy center with once-through cooling. (An artist's conception furnished by the General Electric Company, which does not necessarily endorse the concept.)

Fig. 2 Island site nuclear energy center with wet cooling towers. (An artist's conception furnished by the General Electric Company, which does not necessarily endorse the concept.)

aborted for any reason, the nation would be crippled by a serious shortage of electric generating capacity. The growing need to substitute uranium and coal for the limited supplies of oil and gas means that an interruption of nuclear power growth could have equally serious impacts on all domestic sources of energy. Nuclear energy centers appear to have a potential for reducing the chances of a nuclear option abortion by alleviating the basis for many of the more serious criticisms leveled at nuclear energy and raising public confidence in the process.

BACKGROUND

The concept of nuclear energy centers has been considered in one form or another by various individuals and groups over the past 10 years. In 1965 the United States, Mexico, and the International Atomic Energy Agency initiated a cooperative study to evaluate the feasibility of constructing nuclear energy centers near the United States-Mexico border to produce electricity and desalted water for joint use by the two countries. Results of the study, issued in 1968, concluded that such centers were technically feasible.² Ten thousand thermal megawatts of nuclear energy was considered for the initial phase to meet the near-term anticipated desalted water and electricity needs of the region. It was estimated that future demands to 1995 would require about a fivefold increase in size, or about 50,000 MW(t), at the chosen site.

In 1968 the Oak Ridge National Laboratory (ORNL) conducted a comprehensive study³ on the use of nuclear energy for the generation of electricity and desalted water to serve industrial and agroindustrial complexes. The cases studied involved about 12,000 MW(t), with the expectations that much larger concentrations of energy might be justified in special situations. This study concluded that the economic and technical merits of these agroindustrial power complexes warranted further consideration. The current economic situation enhances the attractiveness of these concepts.

One of the most thorough and detailed studies on energy centers in general was conducted by ORNL in relation to the serious water and food deficiencies of the Middle East.⁴ The study was a detailed technical, economic, and marketing analysis for using, in a specific region of the world, energy, water, agriculture, and industry in an effective combination to help solve some of the difficult problems of the region. The study was not limited to nuclear energy, but much of the information is applicable to nuclear energy centers.

Another specific situation study⁵ was a joint AEC-Department of the Interior and Commonwealth of Puerto Rico evaluation of a small nuclear power plant to provide electricity, desalted water, and process heat for a small diversified industry. It was concluded that the project was technically feasible, but the economic attractiveness was not sufficient to overcome other disadvantages. Again, this particular situation might appear much more favorable in today's environment of high-priced imported oil.

The idea of nuclear energy centers as a reactor siting concept, offering significant advantages over the present diverse siting practices, originated with the former Director of ORNL, Alvin M. Weinberg, who subsequently became the Director of the Energy Research and Development Office of the Federal Energy Administration (FEA). In a paper⁶ presented to the American Nuclear Society (ANS) meeting in 1971, Weinberg suggested nuclear power parks of about 10 reactors each. These parks would have advantages over individual reactor sites of increased safety, less environmental impact, more effective use of sites, and more efficient use of the energy produced. Weinberg expanded on the idea in many other papers, a few of which are referenced.⁷⁻¹⁰ The logic and soundness of the approach stimulated further thinking and studies of the concept by others with indications of growing interest.

In December 1973, Dr. Dixy Lee Ray, then Chairman of the AEC, appointed study groups to evaluate nine proposed demonstration projects to determine their potential for contributing to the Project Independence program. One of the projects for evaluation was nuclear energy centers.

The objective of the study was to determine whether nuclear energy centers offered the potential for meeting the nation's energy requirements faster, cheaper, or more effectively. This involved a critical evaluation of nuclear energy centers, including an identification of the problems, potential advantages, possible need, and recommendations for future actions.

Nuclear energy centers were defined for this study as large concentrations of nuclear power at a single geographical site, with an appropriate combination of related facilities, such as waste-management and storage facilities, fuel-reprocessing plants, fuel-fabrication plants, and energy-using industries. However, the emphasis of the report was on large concentrations of nuclear power. The task-force report¹¹ concluded that the project did not meet the criteria for Project Independence, since nuclear energy

centers are unlikely to make an impact until after 1985. However, the report also concluded that the longer range potential of nuclear energy centers is sufficient to merit further study.

Subsequently FEA Administrator John C. Sawhill suggested¹² nuclear energy centers as a means of dealing with the problems associated with the proliferation of nuclear reactors over hundreds of sites throughout the country within the next 40 to 50 years. He noted that the public appears more concerned about the safety, diversion of nuclear materials, and radioactive waste disposal than the small routine emissions from reactors and their associated systems. Therefore he suggested a strategy to alleviate these concerns—the establishment of nuclear energy centers where the country could concentrate, in large packages, nuclear power reactors and their associated subsystems.

More recently the General Electric Company, under sponsorship of the National Science Foundation, conducted an assessment of energy parks vs. dispersed electric-power-generating facilities. This study includes both fossil and nuclear energy parks. An interim report has been published,¹³ with the full report scheduled for completion in 1975. Figure 3 shows a nuclear park plot plan for an inland site with a cooling pond taken from this report.

Other recent reports include papers presented at the ANS 1974 winter meeting,¹⁴⁻¹⁷ where a session was devoted to nuclear energy centers. The session consisted of four papers, other than this one, on different aspects and plans for nuclear energy centers.

ADVANTAGES OF NUCLEAR ENERGY CENTERS

There are significant potential advantages for clustering nuclear reactors and their subsystems in a limited number of carefully selected sites. Some of these address the more basic criticisms being brought against nuclear energy, and the safety of the nuclear systems would appear to be greatly strengthened by their adoption. The more important of these advantages are listed below:

1. More reliable safeguarding of nuclear materials by increased site security, minimized external transport, and better materials accountability.
2. A higher degree of health and safety protection by concentration of resources, strong centralized management, and reduced shipments of radioactive materials.

3. Lessened impact on the environment as compared with dispersed sites.

4. More efficient use of land and other resources.

5. Establishment of a minimum number of permanent radioactive sites that could be set aside and used indefinitely for this purpose.

6. Provisions for more efficient management, control, and storage of radioactive wastes generated within the center.

7. Faster construction of nuclear power plants by speeding up the licensing process and adopting semifactory-type construction procedures.

8. Possible cost savings by the use of construction patterns applicable to repetitive units over a long period of time.

Willrich and Taylor,¹⁸ among others, point out that the risk for diversion of plutonium from light-water reactors (LWRs) for clandestine uses is the highest at the stage of the fuel cycle where the plutonium has been separated from the spent fuel and prior to its insertion into the reactor. This is especially true for liquid-metal fast breeder reactors (LMFBRs). High-temperature gas-cooled reactors (HTGRs) have a similar situation with the enriched ^{235}U and ^{233}U . However, there are other vulnerable points in the HTGR fuel cycle which are not relevant to this example. Confining the fuel during this stage to nuclear energy centers with internal lines of transportation, better security, and better materials accountability substantially reduces this risk. This advantage applies even though the fuel-recycle facilities are located elsewhere. In this latter case the large concentration of nuclear reactors within the center would facilitate larger, less frequent, but more secure shipments of burned fuel out of the center and refabricated fuel back into the center. Alternatively, reactors colocated with the plutonium fuel-recycle facilities could burn the recovered plutonium, thus keeping even the fabricated plutonium fuel within the centers. It has been suggested that, if the fuel-reprocessing and refabrication facilities were colocated with the nuclear plants, some of the fission products could be left in the fuel, and thus additional safeguards against diversion would be provided. However, this poses serious problems in handling the fuel during the recycle and could involve unacceptable economic penalties.

Nuclear energy centers, because of their immense size, could afford to adopt better control methods and have a cadre of experts for effecting better health and safety measures than would be possible at dispersed sites. Also, this trained cadre and special equipment

Fig. 3 Inland site nuclear energy center with cooling pond. (A preliminary plot plan furnished by the General Electric Company, which does not necessarily endorse the concept.)

could be used immediately to control and clean up any spread of radioactivity resulting from an accident. This should further reduce the small public risks associated with the operation of nuclear reactors.

The regulatory criteria for nuclear energy centers may apply the same chemical and radioactive effluent limitations to a site consisting of a cluster of 10 to 40 reactors as would be applied to a single reactor site. This could place an operational and economic burden on the centers but would result in a reduced environmental impact. Sites for centers are apt to be

more carefully selected and have better characteristics than can be obtained with a much larger number of smaller sites. Thus nuclear energy centers should have less environmental impact per unit of power produced than that from a larger number of smaller capacity sites.

Land management is becoming an increasingly important issue in the development of all our energy-generating and -using facilities. Land availability varies widely from one part of the United States to another. Overall, it is becoming increasingly difficult

for many sections of the country to find suitable and environmentally acceptable sites for the generation and transmission of the electric-power needs. Advance planning and preselection of sites are essential to assure that an adequate number of suitable sites will be available in the future. Development of a small number of carefully selected and planned nuclear energy centers is certainly a rational approach in assuring sound management and use of the available land.

A unique and long-lasting risk associated with nuclear energy is the possibility of land interdiction caused by contamination with radioactive fission products and actinides. Although the property damage and risks to health and safety resulting from any such accident would probably be relatively small, such contaminations could remove land from general public use and access for a long time. The confinement of radioactive operations to relatively few select locations certainly would reduce the amount of land that could be affected in this manner. Land set aside for nuclear energy centers could be dedicated to this purpose in perpetuity, and thus it would be unnecessary to strive for complete decontamination. Also, as old reactors become outmoded, new ones could be built on the same location or on adjacent sites, and thus the areas to be contaminated could be limited and controlled.

One of the major public concerns about nuclear energy is the management and ultimate disposal of the high-level radioactive wastes from the reactors and the fuel-recycle plants. There are several alternatives for treatment and ultimate disposal of these high-level wastes.¹⁹ In each alternative the most vulnerable part of the waste-management chain is handling the waste through the fuel recycle to the waste-solidification stage. Nuclear energy centers should reduce these risks significantly by minimizing external transportation, providing better facilities for handling and treating the wastes, and providing for secure interim storage.

There are several reasons for believing that nuclear energy centers would speed up the construction of nuclear power plants. Front-end actions, such as preparing environmental-impact statements, licensing the plants, and working out the institutional arrangements, may require more time than would be needed for installation of only one or two plants. However, these issues, once decided, would not have to be repeated in the same detail for the second and succeeding phases of the center development. Thus the overall time and effort per unit of power would be less for the complete complex. The same argument applies to public acceptance. Once the local populace has accepted nuclear energy there would be less concern

about future plant additions, as exemplified by installations at Oak Ridge, Tenn., and Hanford, Wash.

The construction of energy centers would consist of a repetitive number of similar units at a single location that favors a semifactory type of operation, having a stable work force for an extended construction period of 15 to 20 years. Adoption of these patterns would surely lead to higher efficiencies, lower costs, and shorter construction times.

The concentration of a larger number of units also encourages the development of different construction modes and practices that tend to lower costs and construction schedules. For example, General Electric Company²⁰ has conceptual ideas of a construction model that offers promise of reduced costs and higher construction efficiencies and appears to be well suited for energy centers. Also, the idea of using shipyard construction techniques for building standardized modules that could be floated on barges to the reactor site, such as visualized by the offshore-system concept,²¹ could be applied in some degree to the construction of nuclear energy centers. In any event, once such centers become standard, they undoubtedly will stimulate many new and productive ideas for improving construction and operating practices.

PROBLEMS TO BE RESOLVED

Studies to date seem to favor the development of nuclear energy centers. However, some of the problems have not been sufficiently defined and analyzed, and at present it is not even possible to establish the ultimate size and composition of such centers. Therefore, before one goes all out on the idea, there are a number of constraints and uncertainties that need to be resolved. Some of the more obvious ones are the following:

1. Technical constraints, such as transmission limitation, climatological effects, concentration of radiological and chemical releases, availability of cooling water, and regional electric-power-growth patterns.

2. Potential for common-mode failures due to natural disturbances, sabotage, acts of war, or nuclear incidents.

3. Institutional, financial, contractual, and organizational arrangements.

4. Legislative, insurance, and tax restraints.

5. Community impacts in areas of population shifts, labor excess, needs for housing, schools, public-service facilities, and transportation.

6. The time required initially for setting up organizational and contractual structures, conducting

the advanced planning, and obtaining approvals for the first plants.

Although all these problems are important and must be given serious consideration, several questions (discussed below) emerge as being critical to the viability of the concept.

What means can be used to reliably transmit the large amounts of electric energy out of the relatively small geographical area? Will the distribution to the consumers of this large amount of electricity place an unacceptable strain on the present transmission grid? The solution to these problems will require consideration of the full range of problems related to the transmission and distribution of electricity. These include, but are by no means limited to, analyzing new as well as conventional means for the transmission of electricity, such as ultrahigh-voltage alternating current, direct current, underground transmission, use of cryogenic means, compressed-gas-insulated lines, and such novel methods as the use of lasers and microwaves and combinations of these.

What will be the climatological effects of such high concentrations of heat and electric energy? Clearly this is an area in which inadequate information exists to make an accurate assessment of the situation. Thus a long-range comprehensive research program needs to be established as early as possible. This program would need to evaluate heat-dissipation methods and effects, possible meteorological changes due to the dissipation of such large amounts of heat (such as the generation of wind storms), and the impacts on the environment in general. The broader ramifications of the climatological problem related to man's production of energy were discussed by Weinberg²² in a recent editorial in *Science*. He presents a strong case for establishing an Institute of Climatology to study the global effects problems of energy generation and consumption. Nuclear energy centers, of course, represent a specific example on a small, local scale of such problems.

Will the potential for common-mode failures be more serious for nuclear energy centers, or will the concentration of resources and talents make them more amenable to control? The "common-mode failures" term as used here includes such natural disasters as hurricanes, tornadoes, earthquakes, and floods or such man-induced failures as acts of war or sabotage. Furthermore, there is not a good understanding of the effect on the energy complex of a radiation accident at any one of the plants. Certain features of nuclear energy centers could work to reduce the adverse consequences of such accidents, but

in other situations such as a nuclear war, nuclear energy centers would seem to be particularly vulnerable. The then dispersed plant site may make the solution a question of great import to the stability of any energy-center concept and needs further study and evaluation that has been given short shrift.

Can the utility industry be organized so that it can accommodate the such centers? Presently, there are about 300 major public and private utilities in the United States and a large number of smaller ones. The establishment of nuclear energy centers will certainly infringe on the independence and on the continued viability of some of the now existing organizations. Those companies in jeopardy would not kindly accept a program that, as a minimum, threatens their corporate identity and as a maximum, threatens their continuation as a separate entity. At the least, the creation of nuclear energy centers would impose institutional, contractual, and management arrangements of a scope and complexity beyond any previous experience of this nature. The utilities' recent financial problems have put great pressures on them to improve efficiencies, reduce high capital-cost capacity, and increase load factors. Energy centers might provide a means for achieving some of these improvements, but the problems in establishing workable and acceptable organizational arrangements are formidable.

Can legislative restraints and social patterns be altered to permit these arrangements which will have profound institutional and social adjustments? Nuclear energy centers will, because of their large electric-power capacities along with associated facilities, serve regional areas that will be large in contrast with areas served by smaller power stations. Cooperative business and industrial arrangements will be required that may not be permissible under present antitrust laws. Regulatory and rate-setting authorities now residing with states may have to be altered. The centers will be located in one state but serve other states. New social patterns will be established, and other patterns will be disrupted or destroyed. Such problems, since they concern people and require political actions, usually are easier to solve in theory than in practice. At the least, their satisfactory solution requires keen insight, good judgment, and a carefully coordinated effort by many parties with different interests.

The high capital investments in energy centers frequently will provide a tax base comparable to, and perhaps larger than, any other private investment within the state. At the same time, much of the product will be transported to adjoining states for consumption. Can the taxes on such an investment be

assessed and allotted in a manner that is equitable and acceptable to all parties?

Who will own and manage the centers? There are many possibilities, one of which is to have a government entity (state or federal) own the land and act as perpetual custodian of the site. This would be desirable and perhaps essential where the semipermanent storage of radioactive waste is involved. Even if the state or federal government should own the land, the management arrangement still poses many difficult problems.

Can enough good sites be found and preserved for this purpose? Congress, in establishing the Nuclear Regulatory Commission, authorized and directed a national survey of nuclear energy-center sites²³ and further directed that a report on the results of this survey be submitted to the Congress and the Council on Environmental Quality not later than 1 year from the enactment of the Act (Oct. 11, 1974). This survey, which is under way, will be a major step in determining the availability of suitable sites and will provide additional valuable information on the feasibility and limitations of nuclear energy centers.

CONCLUSIONS

If the concept of nuclear energy centers proves to be feasible, the result should be that practically all reactors constructed after some established date will be in nuclear energy centers. In light of the manner in which the LWR industry has developed, it probably will not be practical to incorporate all existing reactors into such centers. Moreover, it may be desirable to have some power stations outside the centers to meet local needs, such as peaking power. As noted earlier, the industry could be structured so that plutonium could be burned only in reactors collocated in centers having fuel-reprocessing and refabricating facilities. Thus plants located outside of centers could be limited to uranium-fueled plants. This would confine the handling of plutonium, during those stages in which it would be most vulnerable to diversion, to nuclear energy centers.

Breeder reactors are a special case since they can be located in nuclear energy centers from the beginning if the technical and commercial merits favor such an arrangement. Such a pattern already has been established to a degree, as exemplified by the Liquid-Metal Fast Fuels Test Facility located at the ERDA Hanford, Wash., reservation; the Experimental Breeder Reactor II (EBR-II) located at the Idaho National Engineering Laboratory; and the Clinch River Breeder Reactor and

fuel-recycle demonstration plant planned to be located adjacent to Hohfield National Laboratory. Each of these locations has many of the characteristics and ingredients of a nuclear energy center.

Nuclear energy centers, as previously noted, can take many forms: power only, power plus fuel-recycle facilities, power plus energy-using industries, power plus an agroindustrial complex, or combinations of these. One of the more intriguing new ideas is to establish a large nuclear energy complex in conjunction with a reservoir that can be converted into stored fuel, giving a combination that operates more efficiently than either of the separate entities. For example, nuclear energy might be combined with a large coal bed to produce and use electricity and liquefied or gasified coal products in an efficient combination.

There are many compelling arguments for the national deployment of nuclear energy into energy centers rather than in a large number of dispersed locations. However, the decision to go in this direction will require profound changes in social and industrial institutions and cannot be undertaken without laying the groundwork in many quarters: public, private, industry, labor, state and federal agencies, and Congress. Current evidence appears favorable for moving in this direction, but there remain a number of unresolved questions, and aggressive efforts must be exerted to obtain the answers as soon as possible.

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6. FOREIGN PRACTICE

6.1 A Comparison of Canadian and U.S. Siting Policies

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[Editor's Note: This article was adapted from the Atomic Energy of Canada Limited Report AECL-3156, "A Comparison of Canadian and U. S. Siting Policies," by G. Hake, which was issued in July 1968 as a verbatim record of a presentation at the 1968 joint meeting of the American Nuclear Society and the Canadian Nuclear Association. Because of continuing interest in national and international siting practices, the editors of *Nuclear Safety* felt that Mr. Hake's analysis should be brought to your attention.]

Abstract: The siting conditions that apply to the Canadian Pickering and the U. S. Indian Point 2 and Oyster Creek stations were analyzed in a general way to show that in both countries the approval of these stations has involved three similar conditions: a fifth-order reduction in the total risk, credit for active components of the containment system equivalent to a first- or second-order reduction of risk, and integrated population doses of about 10^6 rems to the thyroid.

It is suggested that a requirement based on a population dose limit is likely to be more easily accommodated than a requirement based on a population distance limit. It is also suggested that, in the practical siting situations likely to arise, the individual dose at the exclusion boundary will govern, and the population center distance criteria as presently defined in the U. S. regulations are therefore unnecessarily restrictive.

Siting practices are not very much different among the major users of nuclear power. However, it would be difficult to support this opinion if one only listened at international siting and safety meetings. A striking example would be to have heard Davis and Robb's paper in 1967 at the International Atomic Energy Agency's Vienna symposium on siting and containment in which they declared that similarity did not even

exist among the sites of reactors already approved in the United States.¹

I have attempted in this article to make a comparative analysis of the Canadian and U. S. siting policies from information that has been made public or disclosed in a somewhat hidden way by the act of approval of a particular station. The information is scanty, and the conclusions were reached by somewhat devious means. Nevertheless, a similarity in approach can be seen to apply if licenses are in fact granted in a consistent manner in the United States and Canada.

In the beginning it was thought that isolation of a site would give sufficient protection to people against the risk a nuclear reactor represented. However, it is now clear that in developed countries the risk can be limited substantially only if some means other than isolation is used and that the hazard to public health can be controlled only by a limited amount through the choice of site. In the United Kingdom, Japan, and much of industrial Europe, the opportunity to use distance even in this way is very restricted. In the United States and Canada, it may be possible to use isolation alone to mitigate a large risk, but the required distance of the station from its load center is not likely to be feasible in economic terms.

Farmer, in the United Kingdom, has often pointed out that, if only one-tenth of the risk represented by the fission-product inventory of a 500-Mw(e) reactor were to be realized, the health prospects of many thousands of people in a metropolis such as London would be affected even if the station were 100 km away.² Approval of current sites seems no longer to depend primarily on the geographical or population

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characteristics of the environment but much more directly on the characteristics of the design of the reactor that is to be put on the site.

It is important to realize that there may be some reactor types which would not be acceptable for power systems simply because the risk they pose to the population cannot be limited sufficiently to the satisfaction of the authorities. The acceptance of the reactor-site combination now depends on whether the regulatory body has sufficient confidence in the safety features of the design and in any other limits that may be put on the operation of the reactor in order to control the consequences of the worst foreseeable accident considered appropriate to the specific case.

In Canada the Pickering station is 29 km from the center of Toronto, and its site is the most sensitive used so far in Canada. The Burlington site³ in the United States would have compared equally with the Pickering site, but the applicant decided not to risk the delays that could arise from consideration of the population-center criterion, which is part of the U. S. requirements. This leaves the Indian Point 2 and Cyster Creek sites,^{4,5} with about half the population density of the Pickering site, as the nearest equivalents to the Pickering site, which has 300,000 people within a 16-km radius. These three sites were examined for a comparison of the siting policies in the United States and Canada. Many of the data used for the analysis of the U. S. case are confirmed by the paper Morris presented at the 1967 siting meeting in Vienna.⁶ For the comparison, I have used the now generally accepted risk of inhalation of ¹³¹I as the predominant hazard. The risk in both countries is that damage to the reactor core could release fission products from the containment normally provided by the design of the fuel itself. The maximum-severity accident is considered in both countries to arise from some breakdown of the fuel-cooling process.

U. S. and Canadian Siting Criteria

The Canadian limit to the hazard has been clearly stated in relation to the maximum-severity accident by the Atomic Energy Control Board as an individual dose of 250 rems to the most sensitive thyroids in the population and an integrated thyroid dose of 1 million rems to the whole population.⁷ In the United States the accident limits are given in 10 CFR 100 (Ref. 8) and variously interpreted in other documents, such as Refs. 6 and 9. The dose limits are 300 rems to the adult thyroid for a 2-hr exposure. The distance at which this dose is accumulated fixes the site exclusion

area. This exclusion area must be surrounded by a low-population zone out to a distance at which the 300-rem dose would be accumulated in a lengthy exposure. Also, the boundaries of population centers involving 25,000 people or more shall not be closer to the reactor than $1\frac{1}{2}$ times the low-population distance. It is not possible to determine from analysis of already approved sites whether a standard for population doses plays a direct role in the evaluation of U. S. nuclear stations.

There are many differences in the data used in the United States and Canada to evaluate the hazard. A meaningful comparison is therefore somewhat difficult to obtain. There are no exposure times formally associated with the Canadian limits, whereas in the United States two exposure times are used; also, the reference thyroids are different in the two countries. Some of the differences can be eliminated by converting the doses to allowable time integrals of concentration.

The allowable dose R to the thyroid in rems from X curies of ¹³¹I is given in Eq. 1, where M is the thyroid mass in grams:

$$R = 12.3 \times 10^7 \frac{X}{M} \quad (1)$$

The thyroid burden of X curies is obtained from

$$X = 0.23 VCt \quad (2)$$

where V = breathing rate, m³/sec

C = concentration of ¹³¹I, curies/m³

t = exposure time, sec

0.23 = commonly used factor for the amount of inhaled iodine that reaches the thyroid

The dose arising from all radioiodines present is twice that due to ¹³¹I alone, and therefore when a limit based on ¹³¹I alone is used, the allowable concentration should be divided by 2. This allowable time integral of concentration, Ct , for ¹³¹I in curie-sec/m³ is given by

$$Ct = 0.177 \times 10^{-7} \frac{M}{V} R \quad (3)$$

Comparisons of M/V ratios used for various individuals are given in Table 1. They are obtained from data of Barry¹⁰ and of Morris and Waterfield.⁶ Differences in assumptions are noticed. The 1-year-old child with the smallest M/V is associated with the lowest allow-

Table 1 Comparison of U. S. and Canadian Values of M/V for Various Individuals

	Weight of thyroid (M), g	Breathing rate (V), $10^{-4} \text{ m}^3/\text{sec}$	M/V ratio $\times 10^4$
Canadian values*			
Adult	25	2.5	10.8
One-year-old child	2.2	0.92	2.39
Child at birth	1.8	0.28	6.41
U. S. values† for an adult			
Short exposure	20	3.47	5.77
Long exposure	20	2.355	8.5

*From Ref. 7.

†From Ref. 6.

able concentration. If the appropriate values for M/V are used in the expression given in Eq. 3, the values in Table 2 for ^{131}I concentration (with allowance for the other iodines) are obtained. Table 2 suggests that the Canadian siting requirement for individual doses, because it is based on a child's thyroid, is five times more severe than the U. S. requirement, but this in the end does not turn out to be so because of other differences in the assumptions, as discussed later.

Table 2 Comparison of Allowable Time Integral of Concentration in the United States and Canada

	Dose (R), rems	Applicable M/V ratio $\times 10^4$	Allowable time integral of concentration, curie-sec/ m^3
Canadian values	250	2.39	0.1
U. S. values			
Short exposure	300	5.77	0.31
Long exposure	300	8.5	0.45

Safety Features Required in Addition to Site Considerations

With the assumption that the predominant hazard is the inhalation of ^{131}I , the maximum potential hazards for the Pickering, Indian Point 2, and Oyster Creek 1 reactors may be represented by the total core inventory of this isotope, as given in Table 3. The total inventory was calculated on the basis of 25 kilocuries of ^{131}I per thermal megawatt at equilibrium.

The limitations applied in an actual case to this maximum hazard are manifold but may be thought of as of two kinds: one kind that occurs outside the

Table 3 Thermal Power and ^{131}I Inventories of Indian Point 2, Oyster Creek, and Pickering Reactors

	Thermal power, Mw	^{131}I inventory, 10^7 curies
Indian Point 2	2760	6.9
Oyster Creek	1600	4
Pickering	1700	4.25

containment system and another kind that occurs inside the containment system. Of the first kind, the dispersion that occurs in the atmosphere predominates. For both countries this dispersion has been represented by some definition of Pasquill F weather.^{6,10} The dispersion is best stated in terms of the volume flow of air past the point of interest by expressing the air mass into which the radioactive release has diffused on a basis that gives a value for the maximum concentration on the center line of the plume. Typical values for Canadian and U. S. definitions of Pasquill F weather are presented in Table 4 and may be seen to be somewhat different.

Table 4 Pasquill F Weather in the United States and Canada

Distance from source, km	Volume flow, $10^3 \text{ m}^3/\text{sec}$	
	Canada	United States
0.5	1.0	0.53
1.0	3.1	1.33
10.0	6.7	2.78

For the kind of limitation that occurs within the containment system, assumptions may be made for the amount of radioactivity released from the core into the containment system. For the U.S. cases for volatiles such as ^{131}I , the quantity is formalized in Ref. 9 as one-fourth the total core inventory. In Canada, for CANDU systems, up to 50% is taken for the maximum case.¹¹

The rate at which this radioactivity could leak from the containment system is the most critical assessment that has to be made in the licensing of reactors in North America, and the rate for a particular reactor depends on the credit given to those safety features which reduce the overpressure in the containment system or modify in some way or another the release. However, for lined concrete pressure vessels, a leak rate of 0.1% of the volume per day at the design pressure has come to be accepted as an achievable standard. The comparison of credit allowed for safety features may

be made against this standard of containment. The allowable concentrations can be converted into allowable total radioactive releases by associating them with the weather dispersion factor appropriate to the distance.

Calculated allowable releases at the exclusion radius and low-population distance are given in Table 5. The additional safety factors determined from the controlling release are given in Table 6. The exclusion radii in this table are taken from the literature, as are the dispersion factors, and this leads to an allowable release by simple multiplication. The low-population allowable release may be seen to be greater than the allowable exclusion-radius release, and so the exclusion-radius release governs. The total inventory multiplied by the release fraction gives the amount of radioactivity free in the containment system. These values provide the first similarity in the siting criteria in that the required reduction factor is about the same for

Table 5 Calculated Allowable Releases at Exclusion Radius and Low-Population Distance

	Station		
	Pickering	Indian Point 2	Oyster Creek
At exclusion radius			
Exclusion radius, km	1.0	0.51	0.4
Weather dispersion flow, m^3/sec	3.1×10^3	5.6×10^2	4×10^2
Allowable time integral of concentration, curie-sec m^3	0.1	0.31	0.31
Allowable release, curies	310	170	122
At low-population distance			
Low-population radius, km		1.02	3.22
Weather dispersion flow, m^3/sec		1.33×10^3	6.08×10^3
Allowable time integral of concentration, curie-sec m^3		0.45	0.45
Allowable release, curies		600	2730

Table 6 Additional Safety Factors Determined from the Controlling Release

	Station		
	Pickering	Indian Point 2	Oyster Creek
Controlling release (from Table 5), curies	310	170	122
Total core inventory of ^{131}I , curies	4.25×10^7	6.9×10^7	4×10^7
Fraction released to containment system	0.5	0.25	0.25
Containment system ^{131}I inventory, curies	2.1×10^7	1.7×10^7	1×10^7
Reduction factor to meet allowable release ^{131}I in 0.1% per day leakage for 2 hr, curies	0.73×10^5	1.0×10^5	0.82×10^5
Additional safety factors on standard containment	6	8	7

the three reactors and is close to 10^5 . If the "standard" containment case is taken as 0.1% per day leakage in 2 hr, the three sites require additional safety features to give a factor of 10 reduction over that provided by the standard containment.

Credits for Safety Features

The doses that result with the credits for safety features which were actually allowed by the licensing bodies for the two U.S. stations were given in the paper by Morris and Waterfield.⁶ In Table 7 these

suppression or washdown sprays. For Oyster Creek the additional safety features are elevated release from the stack and a filter on the release.¹³ In Table 7 it may be seen that the passive containment release accepted by the licensing authority is somewhat better than standard containment release for Indian Point ($P < S$) and somewhat less for Oyster Creek ($P > S$).

The active containment factor was calculated for Pickering from the published figures for the performance of the evacuated building,¹² and it is assumed that these figures were accepted by the Atomic Energy Control Board as achievable. By referring the three

Table 7 Active and Passive Containment Factors

Type of system	Station		
	Pickering	Indian Point 2	Oyster Creek
Allowable release, curies	310	170	122
Design passive leakage factor in 2 hr	0.001	0.0008	0.004
<i>P</i> , passive system release, curies		505	1465
<i>A</i> , active system release, curies	45	74	37
<i>S</i> , standard containment release, curies	1760	1410	830
Passive containment factor, <i>S/P</i>		3	0.5
Active containment factor (safety feature allowance), <i>P/A</i>		7	40
Active containment factor credited above standard containment factor, <i>S/A</i>	40	20	22
Predominant safeguard in reducing release	Evacuated building	Not obvious	Filters and elevated release

doses have been converted to release figures for both a passive system, i.e., one with no credit for devices that have to be activated in some way, and for an active containment system in which some credit, as decided by the licensing body, has been allowed for such dynamic devices as sprays for pressure suppression or washdown. For the Pickering containment system, no attempt was made to decide what might be called a passive system for a proper comparison with the passive systems of the U.S. stations. For Indian Point the passive system is a lined containment vessel with an allowable leakage of 0.1% per day, and for Oyster Creek it is a dry well with suppression pools.

The predominant active component of the Pickering system is the evacuated vessel addition to the reactor buildings, which reduces the time of passive release to minutes.¹² For Indian Point it is not obvious what safety features were credited in addition to the passive system, but they certainly include pressure

reactors back to the 0.1% per day release in 2 hr, as is appropriate to the U.S. case, it may be seen that in fact all three reactors were credited with active containment which gave a similar additional factor; i.e., a factor between 20 and 40 for safety features in addition to standard containment.

Comparison of Population Exposures

To obtain an idea of how population doses compare, it is necessary to relate the three stations to some kind of standard population distribution. For the Pickering site it can be calculated from available figures for metropolitan Toronto and for age-group distributions that, if the individual child's thyroid dose of 250 rems is limiting at 1 km, the population dose to a distance which includes all of metropolitan Toronto is somewhat less than 10^6 rems to the thyroid. Table 8 shows the individual adult doses at 1 km for the two

Table 8 Adult Doses at 1 km from the Three Stations

	Station		
	Pickering	Indian Point 3	Oyster Creek
Exclusion radius, km	1	0.51	0.40
Weather dispersion flow, m ³ /sec	3.1×10^3	5.6×10^2	4×10^2
Adult dose at exclusion radius, rems	55	130	94
Weather dispersion flow at 1 km, m ³ /sec	3.1×10^3	1.33×10^3	1.33×10^3
Adult dose at 1 km, rems	55	55	28

$$\text{Canadian adult dose} = \frac{250 \times 2.4 \times 10^4}{10.8 \times 10^4} = 55 \text{ rems}$$

U.S. stations that correspond to the dose at the exclusion area accepted by the authorities for the active containment system.

The limiting adult dose in Canada at 1 km would be 55 rems, which is the same as that applicable to Indian Point 2. If the U.S. stations were surrounded by the same population distribution as Pickering, the population dose to the thyroid for Indian Point would be about 1×10^6 rems. However, the population around the two U.S. stations is about half that of the Pickering site, so the accepted thyroid dose is about 500,000 rems. If, however, the U.S. exclusion dose had been at the limiting value of 300 rems to the adult thyroid, the population dose would have been about 1×10^6 man rems, and the individual dose at 1 km would have been twice that for Pickering.

Discussion

Despite the similarity in the results of siting practices in the United States and Canada, there is a remarkable difference in the way the requirements are written in the two countries. I was puzzled for a long time as to why the use of a population-center distance in the U.S. regulations appeared to apply a pressure on the siting-approval process that was very much greater than I had noticed for the Canadian situation based on population dose.

The Pickering site, when projected through the station lifetime, seems to be at the upper limit of sensitivity for a site and to be pretty representative of metropolitan siting in North America. I have already said that for such a site, when population thyroid doses are regulated at the generally accepted level of 1×10^6 rems or so, the close-in dose to the individual will govern. The major part of the site-approval process is really to decide what credit can or will be given for

safety features, and until this decision is reached the situation will remain in doubt.

When the close-in individual dose governs, the pressure on the site-approval process relates to the uncertainty in the size of the exclusion radius. This radius has turned out to be about 1 km or so for current stations. It should not be too burdensome economically to cover any uncertainty in such a small distance by a generous margin because sufficient land is often under the control of the applicant for other reasons and, also, effective emergency measures could be quite practical in the small incremental area. If, however, the siting process includes a specific statement on population distance, the situation is quite different. For, then, the uncertainty is the very location of the site. As a case in point, consider a 1500-Mw(t) reactor. In the United States the greatest assurance would be achieved by proceeding according to the minimum requirements of Ref. 9, and for this case the exclusion radius would be 1.4 km and the population center distance 27 km. We might proceed, however, on the basis that we could obtain a credit for a factor of 2 reduction on the release of Ref. 9 for some safety feature, and this would in fact be a very small additional factor.

What are the risks we run in going ahead without approval? If approval is denied, we need to increase the exclusion radius from 0.8 to 1.4 km, a circumstance we may have already wisely accommodated, but we would also need to increase the population-center distance by 11 km from 16 to 27 km. This could only be accommodated by going to an entirely new site, unless we had in the first place located at 27 km. This is not likely, since the object of the exercise was to locate closer to the population center. I am not prepared to debate whether the difference of 11 km is significant in the economics of power generation. I can only say it

appears to be so or there would not be the urge to move closer in.

I suggest that under Canadian rules the remedy against an unfavorable licensing decision can be found at the chosen site, but under U. S. rules the remedy requires removing to a quite different site. Therefore, in the United States, having selected a site, the approval of which depends significantly on credit for safety features, the site remains a possible site only if the safety provisions can be credited. From this situation, I further suggest, comes the great preoccupation of U. S. designers with numbers games and the extensive experimental programs in many areas involving safety features. Among other things, U. S. designers search for small factors in a situation that in the first place was intended to be judged on the basis of biological effects which, in themselves, could not be stated either as to effect or objective to an order of magnitude or so.

Conclusions

One might conclude from the evidence that the population doses that have so far seemed acceptable to the authorities are not very different in the two countries. It is of some interest that a population dose of 2×10^6 man rems is often quoted in France¹⁴ and Japan¹⁵ as an acceptable limit, and a somewhat higher number has been suggested for the United Kingdom when associated with the probability of the maximum accident.¹⁶

For the following reasons, I suggest that the siting policies in the United States and Canada are not markedly dissimilar: (1) both require a fifth-order reduction in the total risk, (2) both allow credit for active components of containment systems equivalent to an additional 10 to 100 reduction of risk over a passive system, and (3) both seem to accept similar population doses. It seems to me it is becoming universally accepted that for any power reactor there must exist sufficient assurance that, under the worst accident conditions foreseeable, the release of activity into the environment shall not produce an effect on the health prospects of the population greater than that due to 100 to 1000 curies of ¹³¹I. This comes about because a release of this size results in reasonable exclusion areas around the plant if the individual doses are to be limited to presently accepted levels and because, with these individual doses, the heaviest foreseeable population concentration around the plant does not give rise to unacceptable population doses in

relation to the probability of occurrence of the accident.

In some further support of this, I can refer to what is being said in the United Kingdom where new siting requirements are being proposed for AGR stations. In the United Kingdom, where very much more power is generated by nuclear means than in any other country, there are no acceptable sites that can be matched to AGR's under the existing siting policies, which are very similar to the U. S. requirements and involve unique distances and limits on population. It is now proposed that no restriction need be put on a site for an AGR beyond a small close-in region of special precautions. The statement has been made that it is a reasonable presumption that this controlled area will be small and, in the United Kingdom, less than 1 mile in radius.¹⁷

The recommendation for this otherwise unrestricted siting of AGR's follows from a sufficient assurance that the individual dose to a child at the edge of the controlled area will not exceed a hundred or so rems and that it is difficult to conceive population distributions which would result in a significant probability that the integrated population dose would exceed 1×10^6 to 2×10^6 rems.

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6.2 German Practices with Respect to Reactor Siting

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[Nucl. Safety, 10(5): 377-379 (September-October 1969)]

Abstract: German policies with respect to reactor siting are similar to those in the United States. Where differences in philosophy exist, the Germans are somewhat more liberal. This is to be expected, since Germany has fewer alternatives to placing reactors in populated areas. As a practical matter, however, the German sites used so far are not more populated than some approved American sites.

Siting policies in several foreign countries are being followed with interest because these countries are expected to adopt, for good reasons, more liberal practices than those of the U. S. regulatory authorities. Among the reasons are (1) the relative scarcity of conventional energy sources and hence a greater need for nuclear energy and (2) a relative scarcity of sites remote from population. Thus there are fewer alternatives to placing nuclear power plants on populated sites.

Siting policies in the United Kingdom were recently the subject of an article in *Nuclear Safety*.¹ Considerable information on the parallel policies in Germany was recently collected in a paper by Birkhofer at the 1968 Winter Meeting of the American Nuclear Society² and in three papers forming a symposium on the "Safety of Site-Independent Nuclear Power Plants" (Sicherheit Standortunabhängiger

Kernkraftwerke) held on Nov. 4 and 5, 1968, in Düsseldorf under the sponsorship of the Institute for Reactor Safety (Institut für Reaktorsicherheit) of the Technical Supervision Committees, Inc. (Technische Überwachungs-Vereine e.V.).³

German considerations of reactor siting and reactor safety in general are quite similar to their American counterparts. The economies of siting near load centers, although not cogent for electric power, are regarded as compelling for process and space-heating steam. The severe accidents discussed are the same as those discussed in the United States—the reactivity accident and the loss-of-coolant accident—and the analysis of these accidents is based on American work supplemented by German investigations on, for example, blowdown phenomena.⁴ The increased emphasis on accident prevention rather than on consequence limitation corresponds to U. S. thinking. The requirements for safeguards in connection with the loss-of-coolant accident (see paper by Jahns et al., Ref. 3, pp. 10 and 11) are very similar to American requirements. Infringement on scenic beauty and historical landmarks by power plants and transmission lines and minimizing such infringement by urban siting

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are familiar topics to the American listener. The large amount of radioactivity emitted from fossil-fueled power plants is also a common topic. As in the United States, there is not complete agreement among experts. Germany uses the limits of 25-rem whole-body exposure and 300-rem thyroid dose in a manner similar to U. S. usage, although they do not have any explicit guidelines corresponding to 10 CFR 100 (Ref. 5).

There are, of course, a few philosophical differences. Schwarzer points out in Ref. 3 that practical sites do not differ much with respect to the population dose in case of an accident and that it therefore does not make sense to approve a reactor for one site and not for another. The small difference in population dose is a result of the assumption of large releases and of the fact that doses above the low level of 150 mrem are considered; consequently the release is felt out to large distances from the reactor. Similar arguments are advanced in the United States, but here a large segment of the opinion holds that the important releases are the smaller ones; the "significant" consequences are then confined to a relatively small area around the reactor, and sites look different.

To a greater extent than in the United States, the nuclear community in Germany believes that a nuclear accident and its causes, prevention, course, and consequences are understood. Accumulation of further experience is not regarded as a prerequisite for urban siting, whereas a view frequently expressed in America holds that further substantial relaxation of siting requirements should not be attempted before at least some plants of the size, power density, and burnup contemplated have been built and have accumulated a few reactor-years of favorable operating record, since this kind of experience could very well point to necessary changes.

Probabilistic accident analysis, so popular in England and Canada and controversial in the United States, also seems to be somewhat controversial in Germany. Reference 2 speaks out against this approach, whereas Schikarski⁶ and Kellermann and Ullrich⁷ favor it. The German confidence in their nuclear plants stems to a great extent from the confidence in rigid preoperational and later periodic inspection. The requirement of inspectability possibly conflicts with underground location, and some Germans are less enthusiastic about this type of construction than the Swedes. Earthquakes are no problem in Germany. Public acceptance is said to be good; but regulatory authorities expressed, in connection with the Kahl reactor, considerable concern about

possible groundwater contamination in case of an accident, and there was some public resistance against Würgassen.⁸

Fig. 1 Population data for the average population of Germany (curve 1) and for approved sites in Germany, United Kingdom, and United States.

There are also design differences in German and American engineered safety features. For instance, the torus containing the pressure-suppression water in designs by General Electric Company is well below the reactor; in German designs, it is level with the core. The rod-ejection grid of General Electric might be replaced in German designs by a nut arrangement.

As a practical matter, the sites so far approved and used in Germany are not substantially more populated than some sites approved in England or America. This does not include the twin reactors proposed by the Badische Anilin- und Sodafabrik AG. The Institute for Reactor Safety reported, essentially approvingly, on this project,⁸ but further regulatory steps are necessary.⁹ Population data for the site are not available to us at present. Approved sites are plotted in Fig. 1, where the population data for Hartlepool (U. K.) and Indian Point (U. S.) are superimposed on the German data from Birkhofer's paper.² For the future, however, Germany has good reasons to be more liberal in its siting practices than the United States.

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7. MEETINGS ON SITING

7.1 IAEA Symposium on Siting Nuclear Facilities

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[*Nucl. Safety*, 5(1): 20-31 (Fall 1963)]

A symposium on "Criteria for Guidance in the Selection of Sites for the Construction of Reactors and Nuclear Research Centers" was held in Bombay, India, March 11-14, 1963. About 120 scientists from 12 countries and five international organizations took part in the four-day symposium, which was sponsored by the International Atomic Energy Agency (IAEA) with the cooperation of the Government of India.

The symposium was in reality an outgrowth of the more limited panel meeting on "Reactor Siting"¹⁵ which was held in Vienna in the fall of 1961 and which was jointly sponsored by IAEA and ISO (International Standards Organization). The Bombay symposium was divided into four sessions that were devoted to a consideration of the various factors that have to be taken into account in the choice of atomic center sites and the criteria that can be applied in the evaluation of particular reactors. For convenience in the discussion of the information presented, a different division of the materials is employed here, and a single paper may be referred to in several sections. This review is also limited to those articles which were generally available at the meeting; it is recognized that some significant material may be missing, although copies of 30 of the 37 or more papers presented at the meeting were examined.

As announced by the IAEA:¹⁶

The Bombay meeting ... [discussed] the criteria for guidance in the selection of sites for the construction of reactors and nuclear research centres. Because of the possible hazards associated with atomic energy operations, the siting of a reactor centre is often a matter of great public interest. In addition to the radiation hazards from the normal

operation of a reactor or arising from a reactor accident, an important consideration relates to the suitability of the site for the disposal of radioactive wastes. Besides, there are major considerations concerning engineering requirements or arising from the desirability of having a nuclear research centre reasonably close to supporting facilities or educational institutes or having a power reactor near enough to the consumers to make it economically viable. The requirements are often complicated by many local factors and are sometimes mutually conflicting. The aim of the Bombay meeting [was] to examine the principal considerations and the criteria which can be applied in dealing with the relevant factors in a specific case.

In general, the considerations involved in siting low-power research reactors and nuclear research centers are the same as those for siting high-production power reactors, but the requirements for the latter are more severe and the relative importance of various factors may change. Although none of the papers presented in this session included all the factors listed below, each of the following factors enters into site selection to some degree:

1. Labor and building costs
2. Living conditions for employees (economic, social, and cultural)
3. Proximity of universities and research centers
4. Capability for future expansion
5. Land costs
6. Utilities costs
7. Transportation costs (roads, railroads, etc.)
8. Proximity of manufacturing firms
9. Site development costs

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10. Public relations
11. Surrounding land use
12. Distance from densely populated areas
13. Interference with other important installations
14. Surrounding population
15. Reactor type and power
16. Containment provisions
17. Floods or earthquakes
18. Meteorology, hydrology, and geology
19. Contamination of water supplies
20. Availability of cooling water
21. Means of waste disposal
22. Capability of storing radioactive materials
23. Local or national zoning
24. Local or national building restrictions

Power-Reactor Siting in the United Kingdom

British philosophy with respect to the siting of power reactors is rather thoroughly described in three papers.¹⁷⁻¹⁹ The first two papers are concerned with many nonnuclear aspects of site selection, and the third is concerned with the evaluation of power-reactor sites. This latter evaluation is the same as that described in the 1961 siting meeting.¹⁵ Of particular interest in the recent paper are the two tables reproduced here as Tables I-3* and I-4.† Table I-3 lists the total doses to the general population which are considered acceptable in major accident conditions in the United Kingdom.

*Table II of Ref. 17.

†Table III of Ref. 17.

Table I-3 TOTAL DOSES TO GENERAL POPULATION ACCEPTABLE IN MAJOR ACCIDENT CONDITIONS IN THE U.K.

1. External radiation	
a. Children up to age of 16 and pregnant women, 20 r	
b. Other persons, 30 r	
2. Beta and gamma dose combined in any superficial tissue	
a. Children up to age of 16, 75 r	
b. Other persons, 150 r	
3. Internal radiation from inhalation and ingestion	
Isotope	Dose to critical organ
I	25 r to the thyroid
Jr ⁹⁰	15 r to the bone
Sr ⁹⁰	1.5 r/year to the bone
Ca ⁴⁵	10 r whole body

These doses are not consistent with the 25-rem whole-body and 300-rem thyroid doses employed in the United States as the basis for fixing the exclusion distance. Table I-4 lists the relative ranges (distances) for limiting exposures from a number of different exposure sources following an accident with a "likely" fission-product release distribution. It is apparent from Table I-4 that ingestion exposures must be given serious consideration in the evaluation of accident consequences.

Examination of the nonnuclear considerations reveals that, in addition to the economic considerations of power needs, water supply, site conditions, employment, population, transportation, etc., the British are especially concerned with the "need to avoid profligacy in the use of natural resources" and a "desire to preserve the amenities of suitable sites." These considerations have resulted in the withdrawal from consideration of over 40% of the land area of England and Wales. "Planning" legislation exists for the national and local coordination of developments that would involve a change of land use or large (>5000 sq ft of floor area) industrial construction, nuclear or otherwise. The selection of suitable sites on this basis for further technical evaluation not only involves conformity to these national plans but also provides for a public inquiry, similar to the Public Hearings²⁰ in the United States, at which objectors have the opportunity to be heard. In pursuing these site policies, the United Kingdom Atomic Energy Authority endeavors to cultivate an informed public opinion to achieve objectivity and realism in assessing nuclear hazards.

The site requirements for nuclear power stations are, collectively, so exacting that it is

Table I-4 RELATIVE LIMIT RANGES FOR A VOLATILE RELEASE

Nature of hazard	Relative range
1. Radiation from cloud	1
2. Radiation from ground deposition: inhalation of strontium	2
3. Inhalation of mixed fission products to lung	4
4. Control of exposed crops (Ca, Sr, deposition)	7
5. Inhalation of iodine	50
6. Iodine on surface crops and in eggs	60
7. Iodine in milk	250

geographically rare to find the desired conjunction of physical characteristics in England or Wales. Although all power-reactor sites chosen so far carry a Class I rating,¹⁵ the number of such sites that appear even remotely acceptable technically for future stations is quite limited; however, relaxation in siting standards may well follow improvements in reactor design and engineered safety features that will result in increased safety.

Power-Reactor Siting in Other Countries

Several other countries, including India, Poland, Sweden, France, and Mexico, were represented by papers that described in varying detail the specific procedures and criteria for siting power reactors. As might be expected, the procedures in none of these other countries approached those of the British or the United States in sophistication, although they frequently contained many common elements.

The practice of the Indian Atomic Energy Establishment is a curious combination of U. S. (Refs. 21 and 22) and British practices, with greater emphasis on the latter. Use is made of the U. S. concepts of activity release from the maximum credible accident and multiple zones around the site; however, the British concept of grading sites on a relative rather than an absolute basis is used, and thus the exposure limits are more comparable with the British limits. On such a basis the Indian Atomic Energy Establishment selected Tarapur, Kalpakkam, and Rawatbhata as sites for the first power stations in western, southern, and north central India, respectively. As has been the experience of nuclear siting authorities in the United Kingdom, the best of several sites as far as accident exposures are concerned must frequently be eliminated for other reasons.

The philosophy of Poland regarding power-reactor site selection²³ was not presented in detail, but it was stated that, in addition to the usual economic and other nonnuclear considerations, there were the following objectives: Power-reactor sites should (1) have no inhabitants in the immediate vicinity, (2) be located in rarely (sparsely) inhabited areas, (3) not be nearer than 20 to 30 km from a big town, which should not lie in the line of prevailing winds, and (4) be downriver from nearby towns. Sites for the first nuclear power stations in Poland

will be in the northeast along the shores of the Big Bug River or the coast of the Baltic Sea.

In Sweden²⁴ the siting problem of interest relative to a power reactor was that for the 65-Mw(t) Agesta reactor, the primary function of which is the production of process heat, rather than electric power generation. Since both the installation costs and the operating costs for heat transmission are high, the reactor was to be sited as close as possible to the consumer area, which was to contain at least 30,000 people. This is in direct contradiction to the general siting objective of large separation distances and is compensated for by the containment features that were provided with the installation. Although the reactor is only 15 km (approximately 9 miles) from the center of Stockholm, it is underground in a steel-lined rock chamber²⁵ that has a prescribed leakage rate of only $2 \times 10^{-4}\%$ of the chamber volume per hour at 1.5 atm overpressure. Since only 10% of the iodine is assumed to be available for leakage, the specified leakage would give an exposure of 25 rem to children of the most sensitive age at a distance of 2 km, which by definition becomes the restricted zone around the reactor. Although the siting policy indicated here resembles that of the United States to the extent that a postulated maximum credible accident is used to determine exposures in the surroundings, the Swedish paper²⁵ indicated concern with the wide range of probabilities associated with several of the factors in the exposure calculation and stated that it would be more consistent and justifiable to employ "meteorological parameters connected with average meteorological conditions as long as the probability for the MCA [could] be considered as very low."

France likewise presented a report²⁶ on the site selection for a specific reactor, the gas-cooled reactor finally located at Chinon. The economic or safety features of the site were discussed in detail, but no claim was made that this siting set a policy. Mexico similarly presented, as a case history, its procedure in selecting a site for a power reactor, the construction of which is quite some time in the future.²⁷ Mexico also presented a site rating index based on the total population exposure.²⁸

Siting Considerations for Nuclear Research Centers

At least six papers were concerned completely or in part with siting problems peculiar to nuclear research centers (Refs. 23, 24, and 29 to 32). Some of the reactors in the various research centers discussed below could be considered power reactors, inasmuch as they have power levels ranging up to 30 Mw(t), but they are discussed here because of the siting factors unique to research centers that were considered. The extent to which many of the previously identified siting considerations enter into the evaluation of the location of a particular nuclear center at a particular site is singularly dependent upon the type and power level of the reactor to be located at the center. Bell and Chicken²⁹ of the United Kingdom identified three classes of research reactor, as described below:

Class I. Reactors that are safe against all accidents to the coolant and in which excess reactivity is positively restricted so that damaging excursions cannot occur are included in Class I. This class of reactor would include "swimming-pool" reactors with plate type fuel elements cooled by natural circulation and generally limited to powers less than 100 kw.

Class II. Reactors that are safe against all loss-of-coolant accidents followed by reactor shutdown and in which reactivity excursions may be prevented by limitation of fuel loading or administrative control of step changes in reactivity are in Class II. This class of reactor would include low-power (≤ 1 Mw) reactors, such as the Argonaut, TRIGA, or AGN reactors.³³

Class III. Reactors that are subject to meltdown following a coolant accident but in which no significant pressure is generated are considered to be Class III. This class of reactor usually employs forced cooling with no significant pressurization of Materials Testing Reactor (MTR) type fuel elements. Although Bell and Chicken²⁹ limited the power level in this class to 3 Mw, there is no apparent reason why higher power reactors with the same safety performance characteristics should not be considered as being in this class.

Bell and Chicken²⁹ concluded that there

[is] no safety reason why research reactors up to 3 Mw power cannot be built on suitable sites near urban districts. By proper design of reactor and

building the release from a research reactor of less than 3 Mw power can be kept to a level at which the inhalation hazard should not extend beyond ~500 yards.

Although not specifically stated, it may be inferred that at ~500 yd the ingestion exposure may be readily controlled and that the "proper design" is capable of limiting I^{131} release in any accident to <10 curies. Obviously the low-powered research reactors are readily capable of meeting this criterion, even when located in a conventional building structure; the General Atomic paper³¹ listed several such reactors located in populated surroundings.

Of particular interest were two papers on studies for siting major nuclear research centers, one by Sweden²⁴ and the other by the Netherlands.³⁰ The two centers are comparable in that large integrated installations envisioned at both sites should be compatible with reactor power levels in the 100-Mw range. The major Swedish considerations, which led to the selection of the site at Studsvik on Tvaren Bay on the Baltic Coast, were the following:

1. Situated on the coast
2. At least 1 km² of ground
3. Sparse population within 3 km
4. No population center with more than 1000 within 15 km
5. A university town within 100 km
6. A town with good schools in the vicinity

Only two places appeared to meet all the criteria, and Studsvik was chosen because of favorable road and railroad connections with Stockholm, which is 108 km away. Although the details were not presented, it appears that the third criterion serves as a limit on the ingestion iodine exposure from the 10% of the iodine inventory released to the building and subsequently filtered and discharged, whereas the 15-km zone was established for the same exposure but on the assumption, no longer considered valid, that the building could be damaged by an accident.

The considerations in the selection of the Reactor Centrum Nederland (RCN) site in Holland were necessarily less stringent because of the much higher population density and land use in Holland as compared with Sweden. The safety of the population in the vicinity of the center

was, however, of primary concern, so the exclusion areas were established around all population centers in which nuclear centers were not to be located, as indicated below:

Population	Exclusion area, km
10,000 to 25,000	10
25,000 to 50,000	12
50,000 to 100,000	14
>100,000	16

In this manner, approximately 75% of the country was excluded. In the remaining areas 30 reference sites were chosen for detailed examination. In this examination the surrounding population was of particular concern. Sites were considered acceptable if no more than 10,000 people lived within 7 km and the number of inhabitants at greater distances did not exceed that determined by the following relation:

$$N = 30R^3$$

where N is the number of inhabitants and R is the distance from the source in kilometers. Of the 30 sites considered, 19 were rejected on the basis of the population analyses, 2 because of the need to safeguard important objects or installations, 26 because of technical requirements, and 11 because of social requirements. The RCN site was selected because it fulfilled all the requirements.

Mexico evaluated six sites for a nuclear research center containing a 1-Mw reactor and a 12-Mw Van de Graaff generator.³² All sites were within 20 to 50 km from Mexico City. The proximity to Mexico City appeared to be a necessary condition because of the requirement for qualified personnel. On the basis of meteorology, seismology, soil mechanics, hydrology, and population distribution, Sulazov and Parres appeared to be the most favorable locations.

Consequences of Calculated and Actual Activity Releases

The ultimate concern in the siting of a nuclear installation is the subsequent exposure of the surrounding population as a consequence of a major activity release. There is not, at the present time, an internationally accepted basis

for the determination of the potential quantity of such a release, and there has been no agreement as to the exposure levels that should be employed for the calculation of control zones. Although the United Kingdom has been concerned with ingestion exposures for some time, particularly since the Windscale incident,³⁴ such considerations have not entered directly into U. S. siting criteria. The ingestion exposures have recently, however, been receiving significant attention. In a comprehensive analysis by Watson and Gamertsfelder³⁵ of Hanford, all the exposure paths from an atmospheric release were examined, as indicated in Table I-5.*

Watson and Gamertsfelder calculated doses for all the exposure paths based on the assumptions listed in Table I-6,† and, on the basis of these assumptions, they calculated doses from (1) exposure to a cloud containing fission products and (2) exposure to ground-deposited fission products as a function of a unit release, as shown in Figs. I-12 and I-13, respectively.‡ The

*Table I of Ref. 35.

†Data taken from Ref. 35.

‡Figs. 5 and 6, respectively, from Ref. 35.

Table I-5 EXPOSURE PATHS—ATMOSPHERIC RELEASE

I. Airborne contaminants
A. External irradiation by:
1. Ambient air contamination*
a. Whole body†
b. Lung (contained air)
c. Skin
B. Internal deposition by:
1. Inhalation*
a. Whole body†
b. Thyroid†
c. Bone
d. Lung and GI tract
II. Deposited contaminants
A. External irradiation by:
1. Contamination of ground,* buildings, and clothing
a. Whole body†
b. Skin
B. Internal deposition by:
1. Ingestion of contaminated fresh leafy vegetables or water
a. Whole body†
b. GI tract†
2. Ingestion of milk* and other food chain products
a. Thyroid†
b. Bone†

*Predominant paths.

†Organs of primary importance.

Table I-6 PERTINENT ASSUMPTIONS IN EXPOSURE CALCULATIONS

A. Fission-product fraction escaping containment	
Noble gases	1.0
Halogens	0.25
Volatile solids	0.15
All others	0.003
B. Atmospheric dispersion and moderate stability: modified diffusion equations	
σ	97 m
b	$0.33 \text{ m}^2 \text{ sec}^{-1}$
b'	$2.5 \times 10^{-4} \text{ sec}^{-1}$
h	0
u	1 m/sec
C. Deposition rates	
$I_{\text{TH}}^{\text{ST}}$	$3.4 \times 10^{-3} \text{ m/sec}$
$S_{\text{e}}^{\text{ST}}, S_{\text{c}}^{\text{ST}}$	$2.2 \times 10^{-4} \text{ m/sec}$
C_{a}^{ST}	$2.2 \times 10^{-4} \text{ m/sec}$
D. Retention of contamination by herbage (uptake through the root system is ignored)	
$I_{\text{TH}}^{\text{ST}}$	100%
$S_{\text{e}}^{\text{ST}}, S_{\text{c}}^{\text{ST}}$	40%
C_{a}^{ST}	20%
E. Amount of herbage cover	
	1 kg/m ²
F. Cow—intake of succulent pasture grass	
Milk production	40 kg/day
Fraction of ingested contaminant secreted in milk at peak output:	
I	10%
Cs	10%
Sr	1%
G. Man—only those assumptions which differ from ICRP recommendations are listed	
Milk consumption rate	1 liter/day
Thyroid mass	2 g
Fractional uptake (ingestion) for a child's thyroid	
	0.3

Fig. 1-12 Dose from exposure to cloud of released fission products vs. decay time.

Fig. 1-13 Dose from ground-deposited fission products vs. decay time.

paper³⁵ stated further that

The relative importance of the cloud and ground-contamination exposure paths depends upon both the release rate and the decay time. It is not feasible here to make a general comparison which examines all possible combinations of these two parameters. A limited comparison is possible by assuming appropriate values for release parameters.... The values selected assume that the exposure to the cloud is the same as the duration of release. The dose from [the] cloud is maximized by ignoring radioactive decay during cloud passage. The estimated doses at two distances for several exposure paths assuming [moderate inversion] conditions are listed in Table [I-7]*....

The dose estimates [of] Table [I-7] are based on the quantity released as measured at one day [for a moderate inversion. However], different decay time [or meteorological conditions] would not change their relative magnitude.

The doses listed in Table I-7 were obtained from analytical calculations and appropriate numerical constants. The validity of the use of the mathematical model to calculate exposures was indicated by an adaptation of a portion of this model to the calculation of the area of ground

*Taken from Table VII of Ref. 35.

†Taken from Table A-IV of Ref. 35.

Table I-7 COMPARISON OF ENVIRONMENTAL PATHS OF EXPOSURE
(MODERATE INVERSION)

Distance, m	Source of exposure*	Organ dose, rem per 1000 curies of mixed fission products released at 1 day			
		Total body (external)	Total body (internal)	Bone	Thyroid
1,000	Cloud	3.2×10^{-2}	7.9×10^{-2}	0.60	11
	Ground	0.19	0.65	0.18	7.8×10^3
100,000	Cloud	9.2×10^{-3}	1.8×10^{-2}	1.4×10^{-3}	1.9×10^{-3}
	Ground	4.4×10^{-4}	1.5×10^{-3}	4.1×10^{-4}	1.3

*The internal dose from ground exposure includes the milk path only.

contamination resulting from a number of actual ground-level releases and comparing these with experimental data. The data are presented in Table I-8.† The paper³⁶ states that

The experimental data are based on Hanford^[36] and the Project Prairie Grass experiments.^[37] The material used in the experiments was zinc sulfide (Hanford) and sulfur dioxide (Project Prairie Grass). The agreement between the model and experimental data is good, especially if it is assumed that zinc sulfide, the material used for the larger areas, has a deposition coefficient equal to $\sim 3 \times 10^{-3}$.

A somewhat complementary report was presented by Horan, Wehmann, and Schmalz³⁸ on the monitoring of radioactive effluents at the National Reactor Testing Station (NRTS). Included in this report was (1) a review of plant-locating practices at the site, where several facilities are grouped together to share common utilities but each group is isolated from other groups, (2) the surrounding population distribution, and (3) the physical characteristics of the site, such as meteorology, hydrology, seismology, and geology.

The atmospheric monitoring program at NRTS was developed to assure that no releases from an operation would result in doses at other sites or to the off-site public that would be in excess of the Federal Radiation Council recommendations.³⁹ Further, as a result of extensive meteorological research over the last six years, reliable calculations of the dispersion of radioactive material in air can now be made for distances of at least 5 miles during most meteorological conditions for releases lasting several hours.

The experience at NRTS was used in the evaluation of data from destructive tests, as

Table I-8 COMPARISON OF MATHEMATICAL MODEL*
WITH EXPERIMENTALLY DETERMINED AREAS

E/Q	$V_A/\bar{u}_0$	Area, sq miles	
		Calculated values for moderately stable conditions	Experimental data for isothermal to stable conditions
10^{-3}	2.2×10^{-4}	0.0025	0.0012-0.0006
	3.4×10^{-3}	0.0020	
10^{-4}	2.2×10^{-4}	0.074	0.012-0.0066
	3.4×10^{-3}	0.029	
10^{-5}	2.2×10^{-4}	2.0	0.14-0.51
	3.4×10^{-3}	0.37	
10^{-1}	2.2×10^{-4}	270	18-31
	3.4×10^{-3}	4.9	

* V_A = deposition velocity, m/sec; $\bar{u}_0$ = average wind speed at ground level, m/sec; $V_A/\bar{u}_0$ = the deposition coefficient, dimensionless; E = time-integrated concentration at ground level, $\mu\text{C}/(\text{m}^2)(\text{m}^3)$; Q = quantity emitted, curies.

well as nuclear incidents.³⁸ In particular, it was concluded from field data obtained following the SPERT-I excursion on Nov. 5, 1962, that:

1. Release of radioactive materials to the atmosphere was less than 1 percent of total fission product inventory.
2. Sutton's diffusion equations fit very closely the field data.
3. The cloud gamma dose was negligible beyond 1500 feet from the reactor.
4. No radioiodines were detected in field samples.
5. Deposition velocity of gross fission products on a grass surface along cloud axis was 0.1-0.2 cm/sec and varied little with distance.
6. Particle size data indicate that along the cloud axis 80 percent of the radioactive material was less than 1.5 microns in size.

In the case of the SL-1 accident, however, the vegetation and milk sampling gave the most sensitive indication of cloud trajectory and low-order contamination.

Extensive measurements have also been carried out at NRTS on the movement of radioactivity in the litho sphere, primarily as a means of monitoring the movement of liquid wastes discharged to the soil. In this program the water level in the soil, as well as the rate and direction of movement of isotopes of concern, has been plotted.

An overall review of the impact of activity releases, both routine and accidental, on agricultural interests surrounding nuclear sites was presented by Wortley of the Food and Agriculture Organization of the United Nations.⁴⁰ Of particular interest in this paper is the following plan for the surveillance and control of waste effluents:

Where dairy cattle are involved then a milk sampling and assaying program will give the required information for adequate protection. This in turn involves the selection of farms likely to be in permanent production suitably oriented around the atomic energy plant and at varying distances from the plant. For example a suitable sampling plan could well include eight to twelve farms located on two rings of the order of one mile and five miles radius from the site. The samples need to be taken at regular intervals, bulked as conditions permit and assayed as frequently as analytical resources will allow. For I-131 with the short half-life of eight days it is necessary to assay at least fortnightly, while for longer lived Sr-90 and Cs-137, analysis of bulked samples at three-monthly intervals would be adequate.

In order to ensure that unexpected hazards have not been overlooked it is advisable to assay soil and herbage for a complete fission product range and alpha-active material such as Pu-239 at least once per year. These samples should be taken from the same farms as those which provide the milk samples. . . . If milk is not involved then a similar procedure should be adopted for the food at next greatest risk.

Wortley⁴⁰ also presented a comparison of the risks from atomic energy operations with those in other fields, including (1) other radiobiological exposures, (2) natural radiation sources, (3) labor statistics from the mining industry, (4) air-pollution (smog) casualties, and (5) chemical-induced mutation. From this comparison it was concluded that

a proper perspective would cause much less public alarm about radiation genetic hazards, noting the relatively small risk involved and the thoroughness of the safety measures which will ensure that this situation can, and will, be maintained.

The general meteorological and waste-disposal problems connected with site selection were treated in two French papers.^{41,42} A paper⁴³ by Lieberman, Pack, and Sirapson of the United States discussed the environmental dispersion and transport systems, specifically meteorology and geohydrology, as they enter into site evaluations.

Effect of Engineered Safety Features on Reactor Siting

Several papers presented descriptions and evaluations of engineered safety features of reactor installations.⁴⁴⁻⁴⁷ Particular emphasis was given to containment provisions that, when coupled with a capability for reducing the concentration of activity within the contained atmosphere, can be extremely effective in reducing the amount of activity released to the atmosphere.

Edwards and O'Neil⁴⁴ of the United Kingdom not only reviewed the essential assumptions involved in the analytical evaluation of various containment systems for water reactors but also presented results on the performance of single- and double-containment systems both with and without iodine filtration. The assumptions used in these calculations are listed below, and the results are presented in Figs. I-14* and I-15:†

Initial release	
I^{131}	10^7 curies
Xe^{135}	2×10^7 curies
Deposited material	
I^{131}	With half-life of 1 and 3 hr (maximum of $6\frac{1}{2}$ half-lives)
Xe^{135}	None
Recycle filtration effects	
I^{131}	Decontamination factor of 3×10^3
I	Filtration rate as specified (2, 2.4, 6, and 24 changes per hour)
Xe	None
Containment features	
Inner volume	10^6 cu ft
Outer volume	Not given
Inner or single containment-vessel leakage	1% / day
Outer containment-vessel leakage	10% / day

*Taken from Fig. 1 of Ref. 44.

†Taken from Fig. 2 of Ref. 44.

SINGLE-SHELL CONTAINMENT VESSEL

Initial Release to Containment Atmosphere: 10^7 Curies of I^{131} , 2×10^7 Curies of Xe^{133}
Assumed Leak Rate: 1% per Day

Curve	Isotope	Deposition Half-Life	Recirculation Cleanup
1	I^{131}	3 hr	None
2	I^{131}	1 hr	None
3	I^{131}	1 hr	6 Changes per Hour
4	I^{131}	3 hr	6 Changes per Hour
5	I^{131}	3 hr	2 Changes per Hour
6	Xe^{133}	3 hr	2 Changes per Hour

Fig. I-14 Performance of single-shell containment vessel.

It is apparent from Figs. I-14 and I-15 that in the cases examined the integrated iodine release was reduced by 1 to 2 orders of magnitude through the use of double containment, depending on the time in question and the assumed deposition and recirculation, etc. It is further apparent that, although deposition and filtration may be effective for reducing iodine concentration, they do not affect the noble gases, which may become controlling in situations where the overall iodine decontamination factor is in excess of 10^3 . A supplementary paper by Rushton and Edwards⁴⁵ presented a more extensive investigation of the performance of double-containment systems with illustrative calculations based on the N.S. Savannah design.

Two U. S. papers^{46,47} indicated a trend toward dependence on some engineered safety features in the siting of power reactors. In particular, the paper by Ergen^{46,48} cited several reactors, some already approved, in which engineered safety features (namely, double containment and iodine filters) have been employed to justify site distances that are substantially less than

those suggested by the existing siting guide.²¹ In view of the incentives for locating power reactors even closer to the load centers, the power industry is quite anxious that sufficient confidence be developed regarding the reliability of engineered safety features to permit even greater reliance on them in the near future. Davis, Schmitz, and Smith⁴⁷ concluded that

The role of engineered safeguards has only indirectly been recognized to date. Our basic concept of the role of engineering safeguards needs considerable clarification. Research programs on nuclear safety are the first place that the recognition of the importance of engineered safeguards should be reflected.

Other Considerations

A paper on waste management⁴⁹ described the implications of waste-disposal considerations with respect to site selection but concluded that, although of economic concern, it was not controlling. Binford⁵⁰ summarized in a straightforward manner the calculation of external and

DOUBLE-SHELL CONTAINMENT VESSEL

Initial Release to Containment Atmosphere: 10^7 Curies of I^{131} , 2×10^7 Curies of Xe^{133}
 Assumed Leak Rates: Inner Shell 1% per Day, Outer Shell 10% per Day

Curve	Isotope	Half-Life	Secondary Filter Rate
1	I^{131}	3 hr	None
2	I^{131}	1 hr	None
3	I^{131}	1 to 3 hr	1 Change per Day
4	Xe^{133}		None or 10% per Day
5	Xe^{133}		1 Change per Day

Fig. I-15 Performance of double-shell containment vessel.

inhalation exposures from stack releases and discussed qualitatively the relative degree of conservatism in several of the elements considered. Smith⁵¹ extended these considerations and advocated a site rating index which was similar to the British scheme but which would include probability factors for wind speed and direction, as well as population distribution.

Another paper by Edwards⁵² reported on a survey of

experimental measurements and theoretical methods of calculating the maximum discharge from pipes containing high temperature water.... Recommendations are made for estimating the limiting flow rates to be expected in three broad divisions; (a) orifices, (b) short pipes ($9L/D < 1$), (c) long pipes.

Menoux, Bovard, and Doury⁵³ presented a review of environmental radioactivity monitoring.

Conclusions

The preceding discussion may have tended to emphasize the differences rather than the similarity of the nuclear siting practices in many countries. It is certainly not surprising that those countries with few nuclear installations and a limited nuclear technology are inclined to adopt conservative siting practices in lieu of a well-developed siting policy. The United States initially located many of its facilities at the isolated NRTS in Idaho; however, nothing has happened that would contradict the conclusion that "most of the plants could have been located in a metropolitan area of any part of the country."⁵⁴

Further, even the superficially different siting criteria of the United States and the United Kingdom, as interpreted by the appropriate licensing authorities, can and have produced siting practices that will become more similar as the

industry grows. In particular, with limited land area in a more densely populated country, the British use relative site ratings. This system permits them to locate uncontained gas-cooled reactors on sites conceivably unsuitable for water-cooled reactors. This is possible because gas-cooled reactors are less subject to severe accidents than water-cooled reactors if the belief is accepted that their maximum activity release is reasonably well defined. In the United States a similar end has been obtained by the containment of water-cooled reactors that could not reasonably be located at remote sites even though this country does have a vast land area. Hence an attempt has been made in the United States to establish criteria of a more absolute nature, such as distance vs. power level. Licensing authorities in both countries are now under pressure to relax siting restrictions on the basis that the nuclear industry has a proven safety record and that engineered safety features which can be provided on nuclear facilities in populated surroundings can more than compensate for the additional risks involved. The reasons most frequently given for proceeding cautiously are that the total number of operating years of nuclear reactors is not great and that it is difficult to prove the reliability of engineered safety features.

In considering the port entrance requirements for the N.S. Savannah,⁵⁴ licensing authorities in the United States were compelled to evaluate the reduction of the distance requirement and the acceptability of engineered safeguards with respect to reactor operation in populated areas. Application of such considerations was inevitable in any event. The extrapolation of similar features to stationary power plants in populated areas must likewise be acceptable on the basis of comparable risks. A trend toward the acceptance of safety features as justification for higher potential, although less probable, population exposures may be evidenced in the siting of recent large power reactors in the United States. Exposures from comparable accidents at these reactors would be much less drastic than in the case of the N.S. Savannah, however. The real question is not whether the location of a power reactor with suitable safety features in a metropolitan area can be justified on the basis of comparable risk with other facilities, but whether AEC would agree that there was sufficient justification to

warrant the greater potential exposure at this time, even though the probability of such exposure might be orders of magnitude less than in existing approved facilities. As stated by Davis,⁶¹

confidence in the engineered safety features is today the most crucial issue in the field of reactor safety and siting. There is a very great need at the present time for large-scale tests on a number of these features to increase confidence in their use.

For the present, those faced with site selection must be guided mostly by the precedents which have been established throughout the world and the attitude of the government agency which must make the final review on a given project. The judgment on the type of containment and the safety features which must be used at alternate sites to meet the government criteria for safety, whether the criteria is [sic] stated or unstated, is a crucial factor in the economic comparison of possible sites on a utility system.

There are several interesting anomalies in site evaluations, at least as performed in the United States and the United Kingdom. Although these anomalies may not appear to be significant when considered in light of variance of the other parameters accepted in such evaluations, each tends to assume more significance when absolute site evaluations are desired. Of special concern in this regard, since it played a role in about half the papers presented, is that of iodine exposure both in terms of acceptable limits and exposure routes. There was some discussion of the iodine dose to the thyroid of an adult; the U. S. site criteria mention a value of 300 rem, whereas in a somewhat different context the United Kingdom uses 25 rem. Edwards and O'Neil⁶⁴ pointed out that the 25-rem dose does not enter into United Kingdom site criteria, as such, but it constitutes an "action limit," above which some remedial action definitely would be intended. They commented that U. S. authorities also would not sit idle until exposures reach 300 rem but would take action at much lower levels. They also expressed the opinion that the advance calculation of iodine exposure is much more uncertain than the advance calculation of whole-body exposure to rare gases, because the latter involves no reliance on retention by fuel elements, plateout, filtration, etc. Hence they felt that iodine exposure limits should be more conservative than

whole-body exposure limits. This is accomplished by assigning both limits the same numerical values, in spite of the lower damage potential of a given number of rem of iodine. On the other hand, by ignoring the iodine ingestion exposure potential, although on the grounds that there is the option of controlling such sources of exposure after an accident, the U. S. siting practice dismisses what could be the most significant consequence of an accident and, incidentally, one which could be somewhat independent of an urban or a rural location. The United Kingdom hazards analysis, however, is specifically concerned with ingestion exposures to iodine which, as indicated by Watson and Gamertsfelder,²⁵ may be orders of magnitude greater than inhalation exposures at the same distance from the point of release.

In closing the symposium, Mr. Pierre Baligand, Deputy Director General of IAEA, said that

the symposium had provided an opportunity for an exchange of information on past experience and the current practice in nuclear site selection as well as for an exchange of views on possible formulation of more widely, or even universally, applicable criteria. At the present stage, the Agency was not trying to frame any rules but rather to stimulate the widest possible exchange of information. The meeting had shown that before universal rules could be framed the main task was to collect the maximum possible information and study individual cases with the help of small groups of experts. IAEA had supplied such expert groups to advise on specific problems at the request of interested parties. [In addition] the meeting had confirmed the impression that the security of nuclear sites from the point of view of public health and safety was at present extremely well supervised all over the world.⁽⁴⁶⁾

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  43. J. A. Lieberman, D. A. Pack, and E. S. Simpson, *Environmental Aspects of Nuclear Facility Site Selection*, presented at the IAEA Symposium on Criteria for Guidance in the Selection of Sites for the Construction of Reactors and Nuclear Research Centres, Bombay, March 11-15, 1963, Paper No. SM-39/16.
  44. A. R. Edwards and R. O'Neil, *A Review of Containment Schemes for Water Reactors*, presented at the IAEA Symposium on Criteria for Guidance in the Selection of Sites for the Construction of Reactors and Nuclear Research Centres, Bombay, March 11-15, 1963, Paper No. SM-39/30.
  45. K. C. Rushton and A. R. Edwards, *Activity Release from a Double-Contained Reactor*, presented at the IAEA Symposium on Criteria for Guidance in the Selection of Sites for the Construction of Reactors and Nuclear Research Centres, Bombay, March 11-15, 1963.
  46. W. K. Ergen, *Site Criteria for Reactors with Multiple Containment*, presented at the IAEA Symposium on Criteria for Guidance in the Selection of Sites for the Construction of Reactors and Nuclear Research Centres, Bombay, March 11-15, 1963, Paper No. 39/22.
  47. W. K. Davis, R. P. Schmitz, and T. H. Smith, *The Role of Criteria in Power Reactor Siting*, presented at the IAEA Symposium on Criteria for Guidance in the Selection of Sites for the Construction of Reactors and Nuclear Research Centres, Bombay, March 11-15, 1963, Paper No. SM-39/22.

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49. W. K. Ergen, Site Criteria for Reactors with Multiple Containment, *Nucl. Safety*, 4(4): 8-14 (June 1963).
  49. J. F. Monstead and J. Bernack, Waste Management Implications in Nuclear Site Selection, presented at the IAEA Symposium on Criteria for Guidance in the Selection of Sites for the Construction of Reactors and Nuclear Research Centres, Bombay, March 11-15, 1963, Paper No. SM-39/46.
  50. F. T. Binford, A Method for Estimating the Upper Bound of the Inhalation Radiation Dose from Reactor Stack Discharge, presented at the IAEA Symposium on Criteria for Guidance in the Selection of Sites for the Construction of Reactors and Nuclear Research Centres, Bombay, March 11-15, 1963, Paper No. SM-39/24.
  51. J. M. Smith, Jr., Locating Nuclear Power Plants to Further Environs Safety, presented at the IAEA Symposium on Criteria for Guidance in the Selection of Sites for the Construction of Reactors and Nuclear Research Centres, Bombay, March 11-15, 1963, Paper No. SM-39/26.
  52. A. R. Edwards, The Discharge of Water-Steam Mixtures from Pipes and Orifices, presented at the IAEA Symposium on Criteria for Guidance in the Selection of Sites for the Construction of Reactors and Nuclear Research Centres, Bombay, March 11-15, 1963.
  53. A. Menoux, P. Bovard, and A. Doury, Arrangements for the Monitoring of Environmental Radioactivity in Conjunction with Safety Surveys of Nuclear Sites, presented at the IAEA Symposium on Criteria for Guidance in the Selection of Sites for the Construction of Reactors and Nuclear Research Centres, Bombay, March 11-15, 1963, Paper No. SM-39/38.
  54. Wm. B. Cottrell, Nuclear Merchant Ship Site Criteria, *Nucl. Safety*, 4(2): 29-33 (December 1962).
  55. IAEA Symposium on Selecting Atom Centre Sites in Bombay, IAEA Press Release PR 63/26.

7.2 The Vienna Symposium on Containment and Siting of Nuclear Power Plants

F. A. Gifford*

[*Nucl. Safety*, 9(2): 273-281 (March-April 1975)]

Abstract: The third in a series of International Atomic Energy Agency symposia on reactor siting was held in Vienna in the spring of 1967. Four principal topics were considered: national practices in reactor siting, reactor siting considerations, containment, and release and transport of pollutants.

The symposium on "Containment and Siting of Nuclear Power Plants" held in Vienna, Apr. 3-7, 1967, was the largest symposium yet organized by the International Atomic Energy Agency (IAEA). Nearly 250 scientists, representing 30 countries as well as ENEA, Euratom, the World Health Organization, and the World Meteorological Organization, took part.

This symposium was the third of a series of IAEA symposia on reactor siting problems; the first was held in Vienna in the fall of 1961 (Ref. 1) and the second, in Bombay, India, in March 1963 (Ref. 2). The large attendance, no less than the spirited nature of the meeting, underlined the continuing concern of the IAEA and the various countries with these important aspects of reactor safety.

The technical presentations involved some 50 papers (Table 1) given in nine sessions and were organized around four principal topics: (1) national practices in reactor siting, (2) reactor siting considerations, (3) containment, and (4) release and transport of pollutants. In view of the large number of papers and the active discussions, individual summaries here would be impractical, and the following is confined to a few brief highlights and impressions on the principal topics.

National practices in reactor siting were articulated, in an interesting series of papers, from several distinct points of view; that is, those of the reactor designer, the government agency engaged in promoting nuclear power, and the government agency responsible for regulation and safety aspects. The presentations reflected some fairly broad areas of international agreement on reactor siting criteria. For example, the radiation dose limit to an individual of 25 rems whole body seems to be universally accepted as a reactor siting and design criterion, and

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Table 1 PAPERS PRESENTED AT SYMPOSIUM ON CONTAINMENT AND SITING
OF NUCLEAR POWER PLANTS

Paper No.*	Paper title	Author	Author's affiliation
National Practices in Reactor Siting			
SM-89/52	Nuclear Plant Siting in the U. S.	W. K. Davis	Bechtel Corporation, San Francisco
SM-89/55	Site Evaluation Procedures in the U. S.	P. A. Morris	USAEC, Washington, D. C.
SM-89/25	The Application of Site Criteria in Japan	H. Osawa	Atomic Energy Bureau, Science & Technology Agency, Tokyo, Japan
SM-89/26	Problems and Efforts in Siting Nuclear Power Plants in Japan	T. Inoue	Nuclear Power Section, Ministry of International Trade & Industry, Tokyo, Japan
SM-89/4	How Reactor Design Is Related to Site Approval and Containment in Canada	G. Hake	Atomic Energy of Canada Ltd., Chalk River, Canada
SM-89/22	Containment and Siting Requirements in Canada	F. C. Boyd	Atomic Energy Control Board, Ottawa, Canada
SM-89/3	Siting of the First Spanish Nuclear Power Stations	F. Pascual S. Noriega A. Sevilla A. Alonso	Junta de Energía Nuclear, Madrid, Spain
SM-89/13	Criteria Applied in the Recent Choice of Site for the First Argentine Nuclear Power Station	E. Evans Morgan	Comisión Nacional de Energía Atómica, Buenos Aires, Argentina
SM-89/14	Site Criteria for Nuclear Power Plants	D. W. Crancher	Atomic Energy Commission, Australia
SM-89/38	Safety and Siting of Nuclear Power Stations in the United Kingdom	C. A. Adams	Central Electricity Generating Board, London, UK
SM-89/41	A Summary of Experience in the Practical Application of Siting Policy in the UK	F. R. Charlesworth W. S. Groomer L. Caribon	Inspectorate of Nuclear Installations, Ministry of Power, London, UK
SM-89/29	Analysis of Criteria and Their Application in Site Selection of Nuclear Power Plants in Yugoslavia	J. Obradovic Lj. Mojovic	AB Atomenergij, Studvik, Sweden; Energo-projekt, Beograd, Yugoslavia
SM-89/57	Quantitative Evaluation of Siting of Nuclear Power Stations from the Safety Point of View	Vincio Serrano C. Carlos Vélez O.	Comisión Nacional de Energía Nuclear, México
SM-89/2	Selection and Assessment of Possible Sites for a Nuclear Power Station to Provide Electricity for the Mexican Grid	F. Sácar Sácar	Comisión Nacional de Energía Nuclear, México
Reactor Siting Considerations			
SM-89/5	Concept of a Standard Site	P. J. Barry	Atomic Energy of Canada Ltd., Chalk River, Ontario, Canada
SM-89/6	Correlation Between Reactor Siting and Containment	O. Knecht	International Atomreaktorbau GmbH., Bensberg/Köln, Federal Republic of Germany
SM-89/51	A Systems Approach to Containment Design in Nuclear Power Plants	S. Levy	General Electric Company, San Jose, Calif.
SM-89/62	Technical Aspects of Choosing a Site for the Saint-Laurent-des-Eaux Nuclear Power Station		Electricité de France, Paris
SM-89/63	Underground Placing of Nuclear Plants and Its Application to the Centrale des Ardennes	P. Grau G. Calberg	SENA
SM-89/64	Radioecological Criteria of the Environment Protection of Nuclear Sites	MM. Bevard Douy	CEA
SM-89/24	Siting Criteria: A New Approach	P. R. Farmer	UKAEA Safeguards Division, AHSB, Risley, Warrington, UK
SM-89/32	The Suitability of Advanced Gas Cooled Reactors for Urban Siting	L. Cave R. E. Holmes	Atomic Power Construction Ltd., Sutton, Surrey, UK
SM-89/20	Siting of Nuclear Power Plants Near Towns	J. Bourgeois J. M. Lavie F. de Vathaire	Centre d'Études Nucléaires de Saclay, France
SM-89/59	Nuclear Plant Siting from the Point of View of Engineering Safety Evaluation	E. Iansiti C. Sennis	Comitato Nazionale per l'Energia Nucleare, Rome, Italy
SM-89/12	The Development of Pressure Differentials Across Containments for Large Water Cooled Power Reactors	H. Karwat	Institut für Less & Regelungstechnik, Technische Hochschule, Munich, Federal Republic of Germany
SM-89/58	Some Examples of the Relationship Between Containment and Other Engineered Safeguards Requirements; Accident Analyses and Site Conditions	W. Vinck H. Maurer	Euratom, Brussels, Belgium
SM-89/8	Comparative Analysis of the Improvement in Safety, Obtained by a Containment and by Other Safety Devices for Various Reactor Types	O. Kellermann H. G. Seipel	Institut für Reaktorsicherheit der TÜV, e.V., Köln, Federal Republic of Germany

*Listed in order of presentation.

Table 1 (Continued)

Paper No.*	Paper title	Author	Author's affiliation
Containment for Particular Reactor Types			
SM-89/1	Containment for Heavy-Water Gas-Cooled Reactors	P. Verstraete D. Lehmann R. Lafitte	Bonnard & Gardel, Lausanne, Switzerland
SM-89/7	Containment of a Large Sodium-Cooled Fast Breeder Reactor	K. Gast W. Schikarski	Kernforschungszentrum, Karlsruhe, Federal Republic of Germany
SM-89/16	Methods of Containment Adopted for the EL-4 Reactor and Projected Heavy-Water, Gas-Cooled Plants	P. Schulhof F. Justin	Centre d'Études Nucléaires de Saclay, France
SM-89/27	Containment Design and Study for the Fast Critical Assembly of JAERI	Y. Iso T. Uchida	Japan Atomic Energy Research Institute, Tokai-Mura, Japan
SM-89/28	Checking of the Leakage Rate from the Leakproof Building of the BP-2 Reactor	F. Leonard P. Boonen M. Carleus G. Rottenberg	Centre d'Étude Nucléaire, Brussels, Belgium
SM-89/31	The Containment of an Advanced Gas-cooled Reactor	L. Cave G. M. Bracewell	Atomic Power Construction Ltd., Sutton, Surrey, UK
Containment for Reactor Types: General Considerations			
SM-89/44	Leakage Rate Tests of the Compartments of the KEMA Suspension Test Reactor	G. van der Lugt J. Kuypers J. P. Rauter	N.V. KEMA, Arnhem, Netherlands
SM-89/38	Steam Generating Heavy Water Reactor: Its Design and Safety Characteristics	F. G. Johnson D. R. Poulter	UKAEA, Reactor Group, Risley, Warrington, UK
SM-89/10	Leakage Rate Testing of Containment Vessels by the Reference Method	R. Boeddege	Kernkraftwerk Lingen GmbH., Dortmund, Federal Republic of Germany
SM-89/21	Recent French Experience in Preventing the Escape of Gaseous Radioactive Wastes at Nuclear Power Plants	P. Candès J. M. Lavie	Centre d'Études Nucléaires de Saclay, France
SM-89/35	The Design of Hinkley Point "B" AGR Pressure Vessel to Avoid the Need for Containment	J. M. Yellowlees K. Robson T. W. Spruce J. O. Tattersall	The Nuclear Power Group, Ltd., Knutsford, Cheshire, UK
SM-89/33	"To Contain, or Not to Contain ..."	P. H. W. Wolff	Nuclear Design and Construction Ltd., Whetstone, Leicester, UK
Release and Transport of Pollutants			
SM-89/15	Trapping of Iodine in Water Pools at 10°C	L. Devell R. Hesbøl E. Bachofner	AB Atomenergi Studsvik, Nyköping, Sweden
SM-89/43	On the Adsorption of Gaseous Methyl Iodide on Active Carbons	G. van der Lugt E. Cornelissens Th. van der Plas J. R. Beattie	N.V. KEMA, Arnhem, Netherlands
SM-89/37	Testing of Reactor Filter Systems in the UKAEA	F. Abbey J. J. Hillary F. J. Billard	UK Atomic Energy Establishment, Risley, Warrington, UK
SM-89/36	Fission Product Removal for Reactors With Vented Containment	A. Barthoux	UKAEA Reactor Group, Windscale, AHSB, Risley, Warrington, UK
SM-89/19	French Practice in the Removal of Radioactive Iodides	C. van Duutselaar	Commissariat à l'Énergie Atomique, Fontenay-aux-Roses, France
SM-89/17	Experimental Studies on Preventing the Escape of Contamination Following a Nuclear Accident	E. C. Watson C. L. Simpson	Centre d'Études Nucléaires de Cadarache, France
SM-89/60	The Influence of Reliability of Reactor Systems on the Probability of Activity Release	H. Bresser W. Schwarzer	Reactor Institute, Delft, Netherlands
SM-89/53	Effect of Wind Variability on Environmental Consequences of Prolonged Releases of Radioactive Contaminants	H. Bresser Ch. Dieck K. H. Lindackers M. Tscherner J. J. Hobbs H. J. Pukarek	Battelle Memorial Institute, Richland, Wash.
SM-89/9	Site Selection for Nuclear Power Plants: Model for the Estimation of Consequences Due to Abnormal Release of Fission Products		Technischer Überwachungs-Verein Rheinland e.V., Institut für Reaktorsicherheit der TÜV e.V., Köln, Federal Republic of Germany
SM-89/11	Site Selection for Nuclear Power Plants: Guides for Estimation of Consequences Due to the Release of Contaminated Air and Water During Normal Operation		Technischer Überwachungs-Verein Rheinland e.V., Köln, Federal Republic of Germany
SM-89/24	The Influence of Thermal Pollution on Nuclear Power Plant Site Selection		Electro-Watt Engineering Services Ltd., Zurich, Switzerland

*Listed in order of presentation.

the population dosage limit of 10^4 man-rems was also referred to in several papers. There are still some differences about what constitutes an acceptable thyroid dosage criterion (Paper SM-89/14), but the area of disagreement is not large.

The more-remote reactor sites have been consumed in some countries; in any event, for economic reasons, sites near population centers have to be evaluated. This point was made in several papers; for example, Papers SM-89/29 and SM-89/39. Thus there was also general recognition that "siting alone is an inadequate means of providing proper safeguards for the public"; that "therefore greater emphasis must be placed on the safety aspect of the design, construction, and operation of nuclear plants" (Paper SM-89/41); and that "it is essential to ensure by safeguards in design and operation that the probability of a large release is reduced to insignificance" (Paper SM-89/39). The problem is, of course, to establish specific criteria in order to implement these principles. A reliability standard for protective devices and containment provisions of 0.003 is in use in Canada; that is, the criterion is that any such device should function as intended 99.7% of the time. Likewise "dangerous faults in the process equipment should not exceed one per three years" (Paper SM-89/22).

Following the sessions on national siting practices, a series of papers dealt with various specific siting problems. Barry (Paper SM-89/5) introduced the concept of a "standard site" and pointed out that, since meteorological differences among sites are not usually large, meteorological considerations could usefully be standardized. A probability-of-failure approach to siting criteria was presented by Farmer (Paper SM-89/34) which led to considerable discussion. In Farmer's method, evaluation of probability of failure leads to a systematic choice of accident paths and definition of the reliability requirements for protective systems. Farmer's interesting thesis was

more fully developed in a concurrent article in *Nuclear Safety*.³ Underground containment (Paper SM-89/63), ecological aspects (Paper SM-89/64), and other interesting siting problems were discussed.

The sessions on reactor containment brought out a number of significant points on containment design. This series of papers covered the containment requirements of various reactor types and experience with various testing methods.

The final session, on release and transport of pollutants, covered such subjects as iodine trapping, the vexing question of methyl iodide removal, fuel-melting experiments, and meteorological factors. With the advent of highly populated sites, these problems have all acquired increased importance, which was reflected by the amount and quality of the results presented.

The proceedings of this symposium will soon be published, and so no attempt was made here to give details of the various papers. The symposium came at a crucial moment for reactor safety, in which the accumulated knowledge of experts is being brought to bear on the difficult question of the siting and safety aspects of locating large reactors in highly populated areas. The papers (and, no less, the discussion and debates that followed them) reflected the continuing concern for the development of safe but economical means of nuclear power production.

References

1. G. W. Parker, IAEA Meeting on Reactor Siting, *Nucl. Safety*, 3(4): 12-17 (June 1962).
2. Wm. B. Cottrell, IAEA Symposium on Siting Nuclear Facilities, *Nucl. Safety*, 5(1): 20-31 (Fall 1963).
3. F. R. Farmer, Reactor Safety and Siting: A Proposed Risk Criterion, *Nucl. Safety*, 8(6): 539-548 (November-December 1967).

7.3 Symposium on Health-Physics Aspects of Nuclear Facility Siting

Charles A. Pelletier*

[*Nucl. Safety*, 12(5): 530-537 (September-October 1971)]

Abstract: The fifth annual midyear symposium of the Health Physics Society was held in Idaho Falls, Idaho, on Nov. 3-6, 1970. Forty-two papers on "Health Physics Aspects of Nuclear Facility Siting" were given in sessions dealing with general site-selection criteria for nuclear power reactors; siting evaluations for large accelerators; fuel, coolant, and fission-product interactions; evaluation of environmental contamination around nuclear facilities; behavior of airborne radioactivity; radionuclides in biological systems; comparison of hazards from internal contamination; radiation standards; and public relations. Rapporteur sessions were held on the environmental impact of operating power plants and fuel-reprocessing plants and on direct radiation from clouds.

The Health Physics Society held a symposium on nuclear facility siting in Idaho Falls on Nov. 3-6, 1970. Siting evaluations for nuclear reactors, fuel-reprocessing plants, and solid-waste-disposal facilities were considered. The program was oriented toward assessment of the behavior of released radioactivity and the consequences for man and his environment. Invited and contributed papers dealt with the quantities and properties of released fission products, measurements and mathematical models of the transport of radioactivity in the environment, evaluation of potential doses and their consequences, philosophies of reactor siting, and siting criteria. The symposium was hosted by the Eastern Idaho Chapter of the Health Physics Society, and about 300 persons attended. Forty-two separate papers were given, and three rapporteur sessions were held that summarized an additional 12 papers.

The keynote address was given by William K. Ergen of the Oak Ridge National Laboratory (ORNL), who showed that certain commonly used site-selection methods yield consistent results with respect to allowable populations surrounding reactor plants. Ergen proposed a site-evaluation method whereby radius factors r^n would be obtained for each person living near a proposed power-reactor site. The value of r would range from 1 to 50 miles, and the negative exponent of r would be equivalent to the n used in Sutton's equation for atmospheric dispersion. The factors would be summed to give a "site characteristic" for use as one measure of the suitability of a site. Ergen showed that, because of existing population patterns

and current attitudes on the economics of electric-power production and distribution, the site characteristic is relatively independent of variations in n from 1.5 to 3.0, which is a reasonable range for most meteorological conditions. Ergen's method is a generalization of previously employed techniques and therefore tends to give results similar to those of other commonly used methods.

FACILITY SITING—POINTS OF VIEW

In the first session, chaired by G. Hoyt Whipple of the University of Michigan, Morton I. Goldman of NUS Corporation described the siting of nuclear facilities in the United States as a three-dimensional "schizoid affair," with the dimensions being "the real world" of the builder and operator, "the somewhat dreamlike world" of the regulator, and the world of "absolute purity and absolute safety" of the vocal public. This was the second but not the last time in this symposium that a psychological interpretation was given to current siting problems. The symposium chairman referred earlier to those working at the interface of nuclear technology and the public as paranoids. At any rate, J. F. Palmer of the Chalk River installation of Atomic Energy of Canada Limited may have presented a cure. In his concluding remarks in the paper following Goldman's, he said that our tendency has been to act like psychiatrists with our "incredibly detailed examination of extreme cases of abnormality" and that "perhaps the time is coming . . . for a much greater emphasis on normality." If it worked, this cure would certainly be preferable to the shock treatment. Goldman emphasized that the professional community should devote much more attention to the "uninformed public and local regulatory opinions with respect to both radiation effects and other environmental factors." His discussion was rich in examples of the kind of information needed to present the professional viewpoint to the public. However, Goldman may have fallen into a trap that not uncommonly arises when defending nuclear power when he made the medical judgment that thyroid cancer is "nonfatal."

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Palmer, in addition to the comments above, presented an analysis of the siting practices in different countries. He observed that there is a lack of firm definition of site-approval policies in some countries, and that policies vary among the countries. However, he observed that there is a measure of agreement on (1) the upper level of credibility of an accident to be accommodated by the design (less than "once every 1,000,000 years or so") and (2) the acceptable level of hazard to the environment from an accident (less than the hazard to the health of a population produced by a release of 100 to 1000 Ci of ^{131}I). In this regard Palmer also pointed out that the Canadian regulation, which prohibits doses greater than 10^4 man-rem per year during normal operation, sets an upper limit to the incidence of leukemia caused by reactor-released radiation of 10 cases per 100 million population per year. He compared this with the natural incidence of 6000 per year per 100 million population. He further observed that there is some agreement on the confidence required in the performance of consequence-limiting devices.

Stig O. W. Bergström of Aktiebolaget Atomenergi of Sweden presented what might be described as a progress report on an official Swedish study of urban siting. The study was prompted by a license application for a large nuclear plant to be operated near the center of Stockholm. He discussed mainly the environmental impact of normal operation. On the basis of their apportionment of allowable dose of 0.3 rem per 30 years (or one generation), he suggested a population dose limit for normal operation of nuclear plants, including the whole fuel cycle, of 4×10^4 man-rem per year for a population of 10 million. By my calculation this leaves about 2×10^6 man-rem per 30 years for accidents. This is to be compared with the Canadian limit of 10^6 man-rem per 30 years given by Palmer in the previous paper and 2.5×10^6 man-rem per 30 years given by Ergen (after F. A. Gifford, Jr.) for the United States in the keynote address. Bergström tentatively concluded that: "No technical problems seem to exist in locating nuclear plants on urban sites with regard to their normal operation." He cited atmospheric releases of fission gases from boiling-water reactors (BWRs) and releases of tritium to water to be used for drinking from pressurized-water reactors (PWRs) as the limiting environmental factors. He also pointed out that the effect of normal operation of a nuclear plant on human health is less by a factor of 10,000 than the effect of a conventional oil-fired plant of the same size. During the discussion, Goldman pointed out that he had arrived at the same factor for

normal operation of a nuclear plant compared with operation of a coal-fired plant of the same size and of the latest design.

SITE SELECTION

The next session, chaired by E. G. Struxness of ORNL, considered in some detail the various factors involved in site selection. D. H. Slade of the U. S. Atomic Energy Commission (AEC) suggested that there was basic meteorological knowledge which was not reflected in the commonly used techniques for obtaining engineering estimates of atmospheric diffusion of radioactivity. He suggested that a research program to develop and test a diffusion model for taking into account spatial and temporal variations might be in order. H. H. Waldron of the U. S. Geological Survey discussed the bases of the criteria being developed by the AEC regulatory staff and its consultants for evaluating a site with respect to seismology and earthquakes. P. J. Carpenter of the U. S. Geological Survey discussed flooding in relation to nuclear plant siting and pointed out that the AEC has informally adopted the "probable maximum flood" concept applied by the Corps of Engineers to the design of major dams whose failure would lead to loss of life. He pointed out that cooling water was to be recycled in some nuclear plants being constructed, and therefore only makeup water would be required from rivers. He suggested that pipelines or canals could be used to transport this relatively small quantity of water so that the plant could be located beyond the flood plain and thus eliminate the risk of a flood. A. P. Jackman of the U. S. Geological Survey concluded this session by describing the results of a study involving the measurement of the dissipation of heat by rivers downstream of seven existing heat sources. The "commonly held concept that thermal discharges from a power plant will affect only 10 to 15 km of a stream" was not verified by these measurements. Jackman pointed out that, in the three sites he considered, in no case did the "heat dissipation approach 50 percent" in a 30-km reach.

SPECIAL SITING EVALUATIONS

In a session on Special Siting Evaluations, chaired by H. Wade Patterson of Lawrence Radiation Laboratory (LRL), R. H. Thomas of LRL presented an analysis which showed it is unlikely that existing and projected high-energy accelerators (energies of several

hundred GeV) would seriously contaminate ground-water supplies.

FUEL, COOLANT, AND FISSION-PRODUCT INTERACTIONS

The next session, which was chaired by W. J. Maack of Idaho Nuclear Corporation, dealt with some of the findings of current investigations of fuel, coolant, and fission-product interactions. D. H. Walker of Idaho Nuclear summarized the status of investigations of fission-product releases from a loss-of-coolant accident for a light-water-cooled reactor. The data he presented from experimental work at ORNL and the National Reactor Testing Station (NRTS) on the fractions of fission products released from melted fuel tend to confirm those used in 10CFR100 and USAEC Report TID-14844 (Refs. 1 and 2). However, Walker pointed out that, with in-core sprays operative, extensive fuel melting would probably not occur, although "at least limited failure of the fuel-element cladding is likely." He described the problem now as being one of determining the fractions of the various radionuclides that would migrate from the fuel to the fuel-pin plenum and thus be available for release when the fuel cladding failed. He said tests have shown that sprays of alkaline solutions can remove iodine "rapidly and effectively" from containment atmospheres.

In his paper on iodine chemistry, J. H. Keller of Idaho Nuclear indicated that hypiodous acid (HIO) is formed best in alkaline solutions at high temperatures. The question is raised therefore of whether the formation of HIO will be the limiting factor on the effectiveness of alkaline sprays. Walker had indicated that the fraction of persistent iodine species may be proportional to the quantity of stable iodine present and had concluded that application of the guidelines of 10CFR100 (TID-14844) to a 1000-MW(e) water-cooled reactor led to a predicted release of 10^6 to 10^5 Ci of ^{131}I . He indicated that in reality, however, the release may be as small as 10 Ci, but additional analytical and experimental data are required before credit can be taken for attenuation factors now considered to be attributable to engineered safety features, such as in-core spray systems and containment spray systems. Walker replied in the affirmative when asked if the noble gases would become limiting if the attenuating factors for radioiodine proved real. During a discussion period later in the symposium, he indicated that the noble-gas dose would become

controlling if the radioiodine release were less than 100 Ci. In answer to a later question, Walker pointed out that the AEC Division of Reactor Licensing is aware of these factors but is not willing to give credit for them because the evidence is not yet sufficiently definite.

Results of studies on the fate of PuO_2 aerosols generated in postulated LMFBR accidents were presented by R. Koontz of Atomics International. He showed good agreement between theoretical results and experimental data which revealed that the rate of decrease in the concentration of an aerosol in a containment atmosphere increases with increasing initial concentration. F. R. Charlesworth of the Ministry of Technology, London, England, summarized the way in which a loss-of-coolant accident occurs in a gas-cooled power reactor and the consequences of such an accident. He primarily discussed the finding that buoyant flow would be the main mechanism of fission-product leakage from the coolant circuit after the circuit had ruptured and depressurized.

There was no skepticism voiced when J. H. Keller of Idaho Nuclear produced further evidence for the existence of volatile HIO. This is in contrast to some scoffs from organic-iodide advocates at the 10th AEC Air Cleaning Conference when the existence of volatile HIO was first proposed. The fact that conditions following a loss-of-coolant accident for a PWR would be ideal for the formation of HIO makes it an important species for study. Walker pointed out in his paper that little is known about its behavior in the natural environment.

Although not directly applicable to reactor accidents, the work discussed by G. M. Kanapilly of the Lovelace Foundation on aerosol formation from solutions of polyvalent metal shed light on the fundamentals of aerosol formation. J. Mishima of Battelle-Northwest concluded the session by presenting the results of a current study on the quantities and characteristics of plutonium made airborne by thermal stress. Forty-six incidents in the nuclear industry involving plutonium were analyzed, and it was found that most of them occurred as the result of explosions during chemical processing. The results of experiments were given in which various chemical and physical forms of plutonium were heated. Fractions of metal made airborne under various conditions of temperature, heating rate, chemical form of the plutonium, etc., range from $5 \times 10^{-5}\%$ to 0.05%.

ENVIRONMENTAL IMPACT OF FUEL-REPROCESSING PLANTS

E. C. Watson of Battelle-Northwest was the rapporteur for a session on the environmental impact of fuel-reprocessing plants. Three papers gave the results of environmental studies by the U. S. Public Health Service around the fuel-reprocessing plant operated by Nuclear Fuel Services, Inc., in western New York. One of the papers was on liquid-waste effluents, and the authors were P. J. Magno and T. C. Reavey; a paper by B. Shleien presented an estimate of radiation dose received by individuals living in the vicinity of a nuclear fuel-reprocessing plant in 1968; and the third paper by J. L. Russell and P. B. Hahn discussed the public-health aspects of ^{129}I from the nuclear power industry. The remaining paper of the session by E. M. Cook and P. G. Stoddard of Argonne National Laboratory summarized the quantities of waste generated at a fuel-recycle complex at the NRTS for servicing a liquid-metal-cooled fast breeder reactor. The Public Health Service studies showed that, although the processing plant was discharging radionuclides, the discharges were controlled so that concentrations in uncontrolled areas were less than those specified in 10CFR20 (Ref. 3). Shleien reported that the dose to a "typical individual" living near the plant did not differ significantly from that due to fallout in other areas of New York. On the other hand, a "maximum individual" (one who substitutes 50 kg of deer meat and 40 kg of fish with the maximum measured radioactivity concentrations for one-half his normal annual diet) would receive in the first year a dose of 38 mrem to the bone from ^{90}Sr and 250 mrem to the whole body from ^{137}Cs and ^{134}Cs . During the discussion it was pointed out that the consumption rates for deer and fish were "hypothetical." Using the specific activity approach to dose estimation, Russell inferred from a measurement of ^{129}I in an aquatic "weed" sample a dose of 290 mrem per year to the human thyroid. A measurement of 210 pCi of ^{129}I per gram of deer thyroid was used in a like manner to infer a dose of 215 mrem. A rather comprehensive program was recommended to determine doses more accurately.

EVALUATION OF ENVIRONMENTAL CONTAMINATION

The session on the evaluation of environmental contamination was chaired by Pamela M. Bryant of the UKAEA Health and Safety Branch. In the first paper

W. E. Bolch of the University of Florida discussed an ecological approach to environmental surveillance around Florida Power Corporation's PWR at Crystal River. He stated that the ecosystem will have been extremely well studied before the plant is operated. With the apparent comprehensiveness of the program and the rigor with which it is being pursued, one almost hopes that the radioactivity from the plant will be measurable in the ecosystem.

J. F. Honstead of Battelle-Northwest Laboratory discussed the methods and results of evaluating population doses resulting from wastes from the Hanford site. He showed that the estimated average whole-body dose decreased from 18 mrem/year in 1964 to 5 mrem/year in 1968 owing to reactor shutdowns. He pointed out the difficulties that were encountered in estimating sizes of the various population groups. Herbert Parker of Battelle-Northwest cautioned later that those interested in obtaining diet information from youths should pick a man to collect data who was not only technically sound but one who was recognized in the community for his youth work, because intense interpersonal contact is apparently involved.

D. E. Gelskey of the University of Michigan described the design and use of an instrument to measure radioactivity in water sediments and on the ground in situ. The field data presented indicate that the instrument is sufficiently sensitive to measure ^{137}Cs in amounts as low as $0.05 \mu\text{Ci}/\text{m}^2$.

H. T. Peterson, Jr., of the Bureau of Radiological Health presented a comprehensive analysis of stable-element and radionuclide reconcentration in aquatic ecosystems. He recommended that measurements of stable elements in both biota and water be included in preoperational surveillance.

J. C. Golden from the University of Michigan concluded the session by outlining a program whereby stable elements would be measured in "monitor organisms" in aquatic pathways to man before a facility was operated. The findings of the program would serve as a basis for "estimating well in advance of operation the potential exposure to man." During the ensuing discussion several attendees were critical of the specific-activity approach to dose estimation. The main criticism was that in several cases the chemical forms of the radionuclide and the stable elements were different and led to different environmental behavior. On the other hand, Peterson pointed out that the same difficulty arises when the simpler "concentration factor" approach is used.

ENVIRONMENTAL IMPACT OF NUCLEAR POWER PLANTS

T. W. Philbin of S. M. Stoller Corporation was the rapporteur of a lively session on the environmental impact of nuclear power plants. He reviewed four papers. One was by A. P. Hull of Brookhaven National Laboratory, who showed whole-body exposure to ^{41}Ar releases from their graphite reactor as the limiting environmental consideration and concluded that such release would probably also be limiting for certain light-water reactors. A paper by R. L. Blanchard and others of the Public Health Service pointed out similar findings around the Dresden boiling-water reactor. A paper by E. A. Belvin and others of the Tennessee Valley Authority gave limits for gaseous releases from the Browns Ferry Nuclear Plant. The fourth paper, by I. S. Bhat and others of the Tarapur Atomic Power Station Colony in India, showed that ^{131}I in the environment was limiting for two boiling-water plants there. Philbin asked some questions to get the discussion going: Why should we monitor anything further than the whole-body exposures and possibly milk because experience is showing that for light-water reactors the noble-gas releases are limiting, the iodine releases are somewhat less important, and the other isotopes are significantly less important? Would water and possibly one critical indicator be sufficient for water releases? Finally, he asked, why monitor the environment at all if the levels are 1 and 2% of the AEC limits? No consensus was reached, and it was probably a blessing that time ran out on the discussion, because in 15 min all the time-worn reasons for performing and not performing an environmental-monitoring program had been mentioned. One exception was the suggestion that perhaps monitoring for long-term buildup should be the responsibility of a federal agency without a credibility gap rather than reactor operators.

BEHAVIOR OF AIRBORNE RADIOACTIVITY

The next session was chaired by D. H. Slade of the AEC and dealt with the behavior of airborne radioactivity. The first paper was given by L. D. Van Vleck of Aerojet-General Corporation (now with Pan American World Airways), who described an experiment in which atmospheric diffusion of "hot puffs" in a stable atmosphere was measured and attempts were made to relate diffusion to measured meteorological parameters. Probability estimates of diffusion were given as a

function of distance, but attempts to relate diffusion to meteorological parameters were largely unsuccessful. The things that D. H. Slade mentioned in his paper earlier, such as wind shear and temporary calm conditions, precluded correlation of diffusion with meteorological variables such as wind speed and the standard deviation of the azimuthal wind direction, σ_θ .

In the next paper L. R. Anspaugh of Lawrence Radiation Laboratory showed evidence for what some consider may be the limiting problem in ground contamination by transuranium elements, such as ^{239}Pu . Two to 4 days after the Schooner cratering detonation, as much as 30% of the original airborne radioactivity concentration of the cloud was measured and attributed to resuspension into the atmosphere. Resuspension was measured by external radiation measurements up to 20 days after initial deposition. A paper by I. A. Singer and Constance Nagle of Brookhaven National Laboratory discussed the concept of wind steadiness as a substitute for the commonly used persistence for estimating downwind concentrations for prolonged releases. Data for Brookhaven, for Peach Bottom Township, Pa., and for Shoreham, N. Y., were presented which showed that, during stable conditions, light winds can persist for a period of up to 4 hr from any direction. However, the chance of persistent winds for a period of as much as 2 days is remote. Wind steadiness for periods greater than 4 hr was shown to be a function of terrain features.

The session was concluded by G. E. Start of the National Oceanic and Atmospheric Administration (NOAA), who is stationed at the National Reactor Testing Station (NRTS) in Idaho. He presented data which showed that the uranine-dye tracer commonly used in determining atmospheric diffusion characteristics may deposit on the ground as readily as ^{131}I gas. His method of measuring uranine-dye particle size (microscopically) was questioned, and the point was made that aerodynamic sizing was preferable. To the extent that the method of uranine-dye generation for NRTS diffusion estimates is the same as that used elsewhere, the practice of taking credit for plume depletion for ^{131}I is open to question.

RADIONUCLIDES IN BIOLOGICAL SYSTEMS

Session 9, chaired by F. L. Rose of Idaho State University, dealt with radionuclides in biological systems. R. J. Chertok of Lawrence Radiation Laboratory

presented a paper in which radioactivity retention and excretion experiments with pigs were described. Debris from the Schooner cratering detonation was used in these experiments. Results were also given for experiments in which debris from three other detonations was used. In two of these experiments, beagles were used. There were variations in retention fractions that Chertok attributed more to differences in chemical and physical forms of the nuclides than to species differences. However, most of the discussion centered on the use of pigs for the experiments. His statement that the gastrointestinal tract of a pig is similar to that of a human was challenged. Some thought the Hanford miniature swine, another breed, would have been a better choice to simulate the gastrointestinal tract of humans.

A. F. Gallegos showed in the next paper that under certain conditions the predominant source of radio-cesium for fish may be that which has deposited in bottom sediments. Therefore the sedimentation of small amounts of ^{137}Cs over long periods of time could be the limiting factor in determining acceptable releases to certain aquatic environments. In the next paper, by J. W. Lentsch and others, 7 years of experience in analyzing the Hudson River estuary for man-made and natural radionuclides was discussed. The findings of this study on fallout of ^{137}Cs supported the cesium findings of Gallegos in the preceding paper. However, Lentsch gave evidence that releases of radio-cesium, postulated to be from Indian Point, might be more biologically available to fish when in the water phase than when in the sediment. Comparative analyses of many radionuclides, including ^{137}Cs , ^{134}Cs , ^{54}Mn , ^{60}Co , and ^{58}Co , showed that ^{137}Cs and ^{134}Cs were the critical radionuclides and that fish consumption was the critical pathway to exposure to liquid waste from Indian Point. During 1969 a person who consumed 30 g of fish per day from an area near Indian Point would have received a whole-body dose of 0.03 mrem. Few environmental measurement programs for a power reactor have been as successful in defining the problem and reaching defensible conclusions.

The next paper, by K. M. Wong of the Woods Hole Oceanographic Institution, showed that plutonium in seawater is concentrated in planktonic, pelagic, and benthic organisms. Wong stated that "plutonium appears now to contribute more than Sr-90 or Cs-137 to the artificial radiation exposure of marine invertebrates." On the other hand, the data for vertebrates show ratios of ^{239}Pu to ^{137}Cs or ^{90}Sr that are considerably lower than those for the invertebrates, which should be comforting to the fish and their

fishermen. The last paper in the session was that of A. F. Jacobson and Herman Cember of Northwestern University, who showed that the ability of *Chlamydomonas*, a unicellular green alga common in polluted and nutrient-rich fresh water, to concentrate ^{144}Ce is increased by a factor of 15 when the temperature is raised from 4 to 22°C.

DIRECT RADIATION FROM CLOUDS

The next session considered the problem of estimating direct radiation from clouds. The rapporteur was M. M. Hendrickson of Battelle-Northwest, who summarized three papers, one of which was his own with D. L. Streng. He discussed the variables that are involved in calculating the cloud dose per se, and also the influence on dose estimation of plateout, aerosol behavior, and multiple containment. He suggested that the best approach to resolving differences in the results of cloud-dose calculations is "to increase communication among environmental dose estimators to the point of obtaining general agreement on basic calculation models." The summary of the paper by C. A. Willis of McDonnell Douglas Astronautics Company suggested that air attenuation considerations could be eliminated from the cloud dose calculation for ground-level releases without causing centerline doses to change by more than a factor of 2 for gamma energies from 0.1 to 10 MeV. Willis indicated in the discussion that the simplified approach was applicable to centerline doses for release heights up to 100 m. This modification would so simplify the equations that hand calculations would be practical and extensive computer calculations would not be necessary. Also, in the discussion, the third author, D. R. Wenzel of Idaho Nuclear, countered by saying that RSAC (a computer code described in his paper) could make a calculation of dose at a point using multienergy groups in 8.7 sec on an IBM 360/75 computer. Single-energy calculations take 0.7 sec. In Wenzel's opinion the limiting factor on the accuracy of the calculation is the assumption of the Gaussian distribution of the cloud, but several symposium attendees defended the use of the Gaussian distribution during the discussion. Singer of Brookhaven National Laboratory indicated that he thought the Gaussian assumption produced concentrations that were accurate to a factor of 2 for ground-level releases of greater than 20 min duration. Bergström suggested that the gamma distribution might be better for elevated releases. One attendee pointed out that the Gaussian distribution "was the only game in town" and that we had to use it. The discussion shifted to the importance

of the often-used "factor of 2." It was generally agreed that the factor of 2 was quite important. One attendee pointed out that a factor of 2 in a dose calculation had recently cost a PWR builder a million dollars.

INTERNAL CONTAMINATION— COMPARING HAZARDS

The next session, chaired by G. L. Voelz of the Los Alamos Scientific Laboratory, dealt with a comparison of hazards from internal contamination. Working with droplets of ^{95}Zr plus ^{95}Nb , Thomas, Walker, and McClellan of the Lovelace Foundation showed that the critical organ in mice changed with the temperature to which the aerosol was heated before inhalation. At the lowest temperature (100°C), the highest doses were received by the skeleton, lung, and liver. At the higher temperatures (600 and 1000°C), the dose was received almost exclusively by the lung. W. W. Schadt presented, with L. Battist, both of the District of Columbia Department of Public Health, the maximum permissible body burdens for 31 radionuclides obtained by using the methods of the Medical Internal Radiation Dose (MIRD) Committee of the Society of Nuclear Medicine. Schadt and Battist concluded that, "except for ^{113}Sn in bone, the ratios for all radionuclides indicate that the ICRP results are conservative by a factor ranging from approximately 1 to 2." During the discussion it was pointed out that the MIRD technique did not include contributions to the critical organ dose from the radionuclides in other parts of the body. Brooks, McClellan, and McKay, also of the Lovelace Foundation, compared the effects of ^{241}Am and ^{144}Ce on the Chinese hamster. Doses of ^{241}Am were found to be 2.5 times more effective than doses of ^{144}Ce in producing early mortality due to bone-marrow damage and 14 times more effective in producing chromosome aberrations in liver cells. The lowest dose to produce a significant increase in aberrations was 13 rads for ^{241}Am and 130 rads for ^{144}Ce . This effect was found to be dose-rate independent down to 7.4 rems/week, and work is continuing on determining the effect at lower dose rates. The last paper in this session was written by O. G. Raabe of the Lovelace Foundation and gave estimates of the relative inhalation hazards for nuclides released from melted reactor fuel. In contrast with other papers given at the symposium, the effects of safety features and containment on the relative release fractions for nuclides were not considered. Nonetheless, ^{131}I was still found to be the limiting radionuclide by a wide margin.

GUIDES, STANDARDS, AND LIMITS

Claire C. Palmiter, the President of the Society, chaired the next session on guides, standards, and limits and had his hands full. The first paper was by Pamela M. Bryant, in which she discussed the derivation and application of "Derived Working Limits" (DWL) for normal releases from nuclear facilities in England and "Emergency Reference Levels" (ERL) for emergency preplanning and guidance for executive decisions on emergency actions. Miss Bryant calculated a DWL for ^{129}I based on the specific activity approach and an air concentration of stable iodine of $0.1 \mu\text{g}/\text{m}^3$. The DWL was 1.2 Ci per day for a 120-m stack. This is equivalent to about 50% of the world's annual release of ^{129}I to the environment estimated for the year 2060 according to figures given earlier by Russell (9×10^6 Ci per year times 0.01 due to removal). It is also interesting to note that in this case the specific activity approach leads to a higher permissible release than would be the case if just the nuclide alone were considered. Miss Bryant's DWL for ^{131}I was 0.65 Ci per day. The ERLs, given in terms of dose, were 25 rems to the thyroid, 10 rems to the whole body, 15 rems to the lung, and 1.5 rad per year to the mineral bone from ^{90}Sr . These doses were apparently set on the basis of an acceptable risk, although the rationale was not given by Miss Bryant in her paper. In the next paper B. J. Mann of the University of California at Berkeley compared these ERLs with the AEC's siting criteria set forth in 10CFR100 (Ref. 1). His main thrust seemed to be that the risk of thyroid cancer from the 300-rem criterion for thyroid exposures was higher than the risk of leukemia from the 25-rem criterion for whole-body exposure, and therefore the criteria were inconsistent. During the discussion it was brought out that the limits in 10CFR100 are not exposure limits but, rather, design criteria to establish population distances. An attendee from industry said that in reality the AEC does not allow potential reactor operators to design these criteria.

H. F. Soule of AEC's new Division of Waste and Scrap Management described the basis for criteria to limit the quantities of solid waste that can be buried at selected locations other than the salt-mine repository now under development. He recommended that solid wastes containing up to 10 nCi of ^{239}Pu per gram "continue to be buried at carefully selected sites, as at present." During the discussion Dr. Morgan of ORNL cautioned that it be stressed that the 10 nCi per gram value is allowable only at a few carefully selected sites, since not to stress this point would provide ammunition for the critics of atomic energy development.

Soule's presentation provided a sharp contrast for the next paper by C. R. Miller and M. Inclan-Suarez of the Georgia Institute of Technology. Miller played an interesting word game with the recommendations of the International Commission on Radiological Protection (ICRP) and the National Council on Radiation Protection (NCRP). An engineering interpretation of biological data led Miller to conclude that "present MPD and MPC levels will be found too high" and "there is no 'safety factor' directly incorporated into the determination of MPD or MPC values."

C. D. Corbit of Douglas United Nuclear tried to put radiation in perspective. On the basis of the operating experience of 11 nuclear power reactors, Corbit estimated that the Hanford site could house 265,625 such reactors without exposing the surrounding population to more than 17 mrems per year. When asked in the discussion what he thought the chances were of having maximum credible accidents at these reactors, he replied "it's statistically possible, though not pragmatically probable, that a man can throw a golf ball over the Empire State Building." Corbit went to even greater lengths than Goldman to provide understandable analogies for putting radiation in perspective.

FACING A CONCERNED PUBLIC

The last session was chaired by E. Dale Trout of Oregon State University. M. E. Wrenn of New York University discussed the work of the Greater New York Chapter of the Health Physics Society in bringing the real issues of nuclear power before the public. P. M. Lorio of Columbia University followed with a description of the legalities of getting a research or power-

reactor license and then with a description of the groundwork his experience has shown must be done to get such a license in an urban area. He stated that the public must first be convinced of the benefits of the reactor before the legal procedure has a chance of ending in a license. J. E. Martin concluded the symposium by describing the current role of the U. S. Public Health Service in the controversy over nuclear power as unique. According to him, "the success or failure of nuclear power may well depend on the availability of credible, nonbiased, reliable data, such as that provided by health agency programs." During the discussion of this paper, one attendee expressed doubt whether the public image of an agency could remain that of an unbiased observer once it had taken a stand on an issue.

PROCEEDINGS

The proceedings of the symposium are available at a cost of \$13.00 plus postage from B. R. Baldwin, Publication Chairman, P. O. Box 2431, Idaho Falls, Idaho 83401.

REFERENCES

1. *Code of Federal Regulations*, Title 10, Part 100, Reactor Site Criteria, June 1, 1962.
2. J. Di Nunno et al., Calculation of Distance Factors for Power and Test Reactor Sites, USAEC Report TID-14844, Mar. 23, 1962.
3. *Code of Federal Regulations*, Title 10, Part 20, Standards for Protection Against Radiation.

7.4 Review of ANS Topical Meeting on Nuclear Power-Plant Siting

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Abstract: The American Nuclear Society Topical Meeting on Nuclear Power Plant Siting, held in Portland, Oreg., Aug. 25-28, 1974, included sessions devoted to the government and siting, methodologies of site selection, special and technical issues, utility siting experience, and social and public acceptance issues. Highlights of the meeting included speeches by AEC Chairman Dixy Lee Ray, Michael McCloskey of the Sierra Club, and Saul Levner of the AEC probabilistic risk study team. Utilities' problems and practices in siting nuclear power plants were discussed by four American, as well as representatives of French and Swiss, utilities. The lack of wholly suitable sites and concern with public acceptance were common themes addressed in widely different manners. Institutional changes in siting procedures and approaches were suggested in both the first session on governmental roles and the final session on social issues.

The American Nuclear Society (ANS) Topical Meeting on Nuclear Power Plant Siting, held in Portland, Oreg., Aug. 25-28, 1974, included sessions devoted to the government and siting, methodologies of site selection, special and technical issues, utility siting experience, and social and public acceptance issues.¹ Both skeptical and optimistic views of nuclear power were presented, beginning with AEC Chairman Dixy Lee Ray's optimistic forecasts for nuclear power and her review of problems and prospects.

Chairman Ray's keynote speech emphasized the need for reasonable, accurate public information on nuclear power, hailing the Rasmussen reactor safety study² on accident probabilities and a paper entitled "The Nuclear Debate: A Call to Reason"³ as major contributions to that debate. She warned that it is not in the public interest to continue harping on the hazards of nuclear energy, just as it is foolish and ignorant to ignore such hazards and that the public has a right to expect responsible precautions.

Among the AEC's accomplishments in the past year as listed by Chairman Ray are (1) improved public participation, (2) limited work authorization, and (3) issuance of 11 operating licenses, leaving no backlog. Future AEC emphases will include investigation of standardized plant design, power-park concepts, and long-distance transmission needed to permit power

parks. She pledged more openness from within the AEC while lightly chastising those who would "cage knowledge" and prevent its use for the benefit of mankind.

A "skeptical view of nuclear power" was provided by luncheon speaker Michael McCloskey of the Sierra Club. McCloskey cited the many reasons the Sierra Club wanted to believe in nuclear power and then related the group's increasing concern over (1) waste-storage problems "which the present beneficiaries cannot control," (2) premature promotion of large-scale development of nuclear technology, (3) the huge capital demands of the nuclear power industry, and (4) massive research and development subsidies that heavily favor the nuclear alternative over other energy sources.

McCloskey questioned "whether we can afford to gamble so much" since prodigious sums of capital are involved, as are people's lives and mankind's gene pool. He further stated: "We are forcing words of acceptance on all species and all people for all time." He charged that the AEC has rarely acted on problems until forced and questioned its delays on adopting safety practices, derating certain plants, or acting on solid-waste needs and safeguards. Then he asked: "With so much laxity, how can the public know the risks it is assuming?" He urged the AEC to convince its critics that answers can be found to these outstanding problems and suggested measures to improve credibility, such as standardizing plant designs, building smaller plants, avoiding shoreline sites, and supporting efforts to separate long-lived actinides from other nuclear wastes. He said: "Industry is risking everything if it continues business as usual, and credibility is dropping so fast that soon no answers will be trusted."

The meeting consisted of six consecutive sessions of four to eight papers each. The complete agenda for the meeting is presented in Table 1. Here each session is discussed separately, and the discussions are followed by the reviewers' conclusions.

*Oak Ridge National Laboratory.

Table 1 Agenda of the ANS Topical Meeting on Nuclear Power-Plant Siting**Plenary Session****Welcome and Introduction:** Ralph Davis, President, Puget Sound Power & Light Company**Keynote Address:** Dixy Lee Ray, Chairman, U. S. Atomic Energy Commission**Session 1 The Government and Siting****Chairman:** P. Stern (Northeast Utilities Service Company)

- 1.1 The Government and Nuclear Power Plant Siting, Jared J. Davis (U. S. Atomic Energy Commission), invited paper
- 1.2 The Thing is the Societal Impact, Lawrence B. Bradley (Washington State Department of Commerce and Economic Development), invited paper
- 1.3 Site Acceptability and Power Availability—Needed Institutional Changes, Joel E. Haggard (Houghton, Clark, Coughlin & Riley)
- 1.4 Practice and Experience of Safety in Nuclear Plant Siting in France, P. Candes (Commissariat à l'Énergie Atomique, France), invited paper

Session 2 Methodology of Siting**Chairman:** D. Muller (U. S. Atomic Energy Commission)

- 2.1 Selected Environmental Considerations: The Guide to Siting of Nuclear Power Plants, Jan A. Norris (U. S. Atomic Energy Commission)
- 2.2 A Systematic Approach To Evaluate Sites for Nuclear Power Plants, J. A. Fischer and R. Ahmed (Dames & Moore)
- 2.3 The Methodology and Criteria Used for a Nuclear Power Plant Siting Study, L. J. West (Dames & Moore) and Ken Dyar (Public Power Council)
- 2.4 The Environmental Assessment Matrix as a Site Selection Tool—A Case Study, Thomas A. Tamblyn and Edward A. Odenborg (Bechtel Power Corp.)
- 2.5 Numerical Rating of Environmental Impacts for Power Plant Siting, Haldon L. Smith and William L. Heilman (Commonwealth Associates, Inc.)
- 2.6 Quantification of Siting Factors for Rational Selection of Alternate Sites, Lawrence P. Beer (Environmental Science and Engineering, Inc.)
- 2.7 Systematic Methods for Power Plant Siting Studies, Neil M. Geitner and Kenneth R. Broome (Gilbert Associates, Inc.)
- 2.8 Probabilistic Methods in Reactor Risk Estimation and Siting, Saul Levine (AEC) and N. Rasmussen (Massachusetts Institute of Technology), invited paper

Session 3 Technical Issues in Siting**Chairman:** C. Gill (Bechtel Power Corp.)

- 3.1 Rationale of Reactor Site Selection for Public Safety, J. R. Beattie (United Kingdom Atomic Energy Authority, England)
- 3.2 A Technique for Consideration of Population in Site Comparison, J. Kohler, A. Kenneke, and B. Grimes (U. S. Atomic Energy Commission)
- 3.3 Siting Considerations for Radioactivity in Reactor Effluents During Normal Operation, J. M. Graf and P. O. Strom (U. S. Atomic Energy Commission)
- 3.4 The Assessment of the Radiological Environmental Impact of Nuclear Reactors, M. A. Parsons and J. Kastner (U. S. Atomic Energy Commission)
- 3.5 Safety Evaluations of the HTGR as Related to 10 CFR 100 Siting Requirements, T. R. Moffette (General Atomic Company)
- 3.6 Sensitivity of Diffusion Estimates to Meteorological Data Recovery and Accuracy, Keith Woodward (Pickard, Lowe and Associates, Inc.)

(Table continues on the next page.)

Table 1 (Continued)

Session 4 Special Siting Issues**Chairman: G. Jacobsen (Puget Sound Power and Light Company)**

- 4.1 Thermal Discharge from Power Generation at Inland and Coastal Sites, Russell M. Miser (Pacific Gas & Electric Company)
- 4.2 An Agricultural Cost/Benefit Analysis Favoring Cooling Ponds for Nuclear Power Stations, R. A. Cushman (U. S. Atomic Energy Commission)
- 4.3 Siting Alternatives for Floating Nuclear Plants, Fredric E. Bontack (Offshore Power Systems)
- 4.4 Underground Nuclear Plant Siting—A Technical and Safety Assessment, P. L. Doan and J. H. Crowley (United Engineers & Contractors Inc.) and D. M. McCreath (Acres American Inc.)
- 4.5 Hazards to Nuclear Plants from Surface Traffic Accidents, Karl Hornyk (Oregon State University)
- 4.6 Siting Envelope for Standardized Plants, W. D. Gilbert and J. F. Quirk (General Electric Company, San Jose)

Session 5 Siting Experience**Chairman: J. L. Williams (Portland General Electric Company)**

- 5.1 Site Selection and Evaluation of Alternatives by U. S. Utilities, A. David Rossin (Commonwealth Edison Company) and Barry L. Nichols (Atomic Industrial Forum), invited paper
- 5.2 Site Selection: A Continuing Process at Commonwealth Edison, K. D. Leitzell, R. A. Nantz, A. David Rossin, and N. E. Wandke (Commonwealth Edison Company), invited paper
- 5.3 Nuclear Power Plant Siting Considerations and Experiences, P. M. Krishau and R. P. Douglas (Public Service Electric & Gas Company of New Jersey), invited paper
- 5.4 Nuclear Power Plant Siting in Southern California, David J. Fogarty (Southern California Edison Company), invited paper
- 5.5 The Douglas Point Decision: A Case Study, John J. Scoville (Potomac Electric Power Company)
- 5.6 A Regional Siting Survey for Thermal Power Plant Sites in the State of Ohio, Michael L. Elkins and Joseph J. DiNunno (NUS Corporation) and Paul V. Morgan (NUS Corporation, Cyrus Wm. Rice Division)
- 5.7 Siting of Nuclear Power Plants in Switzerland, R. Véya, Baden, Switzerland
- 5.8 Siting of Power Stations in France, A. Bellin and P. Toureau (Electricite de France)

Session 6 Siting and Social Issues**Chairman: R. Taliman (Bonneville Power Administration)**

- 6.1 A Method for Combining Societal and Technical Judgments in Environmental Decision Making, John B. Burnham (Battelle Northwest Laboratories) and Stan M. Nealy and William S. Maynard (Battelle Human Affairs Research Center, Seattle)
- 6.2 A Method for the Quantification of Aesthetic Values for Environmental Decision Making, Grant R. Jones and Brian A. Gray (Jones & Jones) and John B. Burnham (Battelle Northwest Laboratories)
- 6.3 Social Considerations in Nuclear Power Plant Siting: Choosing the Least-Social-Cost Site, Elizabeth Preelle (Oak Ridge National Laboratory)
- 6.4 A Methodology for Evaluating Community Acceptance of Power Plants, E. W. Sucov (Westinghouse Research Laboratories) and C. K. Liang (University of Hong Kong)
- 6.5 The Impact of Zoning on Nuclear Plant Siting, Michael V. Stimac (Puget Sound Power & Light Company)

SESSION 1: THE GOVERNMENT AND SITING

Present and suggested future government roles in power-plant siting were the focus of Session 1. Suggested modifications of federal and state actions and interactions were discussed.

According to Davis (Paper 1.1), we must break away from the present case-by-case consideration of nuclear sites if licensing time is to be reduced. The best prospect for reducing licensing time is to have a bank of predesignated sites for standardized design plants. To further encourage state and regional bodies to become more active in land-use planning, an AEC-sponsored conference on the designated-siting program will be held in January 1975. Questions to Davis indicated concern as to how predesignation site studies will be financed and about the feasibility of likelihood of the requisite regional planning by the states.

Two major papers dealt with social and institutional considerations, both using the pioneering Washington State thermal power-plant siting law as a discussion base. Lawrence Bradley, Washington State Department of Commerce and Economic Development, related experiences and considerations from implementing the law over its 4-year life in Paper 1.2. He noted that such laws are vital and significant because social impacts are "more provincial than national" and because local people often lack trust in the performance both of the utilities and of the federal government. Bradley sees that law as a courageous local recognition of these problems and a determination that future growth must be orderly.

The recent Skagit County contract zoning arrangement with Puget Sound Power & Light Company can be attributed to the protection offered by the siting law and to the county's belief that it has "a right to know" details and to have some control over events affecting its future. In recognition of the growing importance of social impacts in siting, a proposed and introduced amendment to the siting law would require a social-economic impact report.

In the second paper on institutional factors in siting (Paper 1.3), Haggard suggested institutional changes needed to permit resolution of competing energy-environmental goals while permitting fulfillment of national energy self-sufficiency objectives. Also using the Washington State thermal power-plant siting council as a model, he recommended creation of a comparable federal siting council to consolidate multiple federal-agency involvements in site approval.

Joint action by federal-state siting councils would permit unified processing of a single environmental impact statement, approval or disapproval of specific sites, and reduction of time-consuming, uncoordinated applications. Only four states now provide consolidated siting procedures (New York, California, Washington, and Maryland). Haggard called for more comprehensive criteria than those in the recent AEC general environmental siting guidelines, including societal, economic, and state land-use planning factors and goals.

Most of Paper 1.4 by Candes on the French experience and plans for siting nuclear power plants dealt with the assumptions and proposed criteria for siting and presented little that was new or different from the more advanced American experience. The French Ministry for Industry is responsible for nuclear reactor safety and has only recently received authority and completed procedures for authorizing plant construction and startup. Possible safety-procedure requirements are being reviewed.

SESSION 2: METHODOLOGY OF SITING

This session concerned the selection of the best site or sites from a number of alternatives. Environmental factors were dominant in these selection methods, largely because sites with obvious safety problems are usually eliminated in a preliminary screening. During the question session following his paper, Fischer (Paper 2.2) remarked that safety considerations are usually a "go, no-go" proposition, in that sites with known hazards, such as an active seismic fault, or sites located near a major airport are simply eliminated from further consideration. Smith (Paper 2.5) expressed much the same idea in that he observed that the site-selection process occurs in three stages: (1) the selection of candidate areas, (2) the preliminary evaluation of sites in the candidate areas to determine a set of suitable sites, and (3) the selection of the final site or sites from the selected set. In this process, safety considerations come first, in stage 1 whenever possible. For example, an area of high seismic activity is eliminated in stage 1.

Several of these papers, including those by Fischer (Paper 2.2), Tamblyn (Paper 2.4), Smith (Paper 2.5), and Beer (Paper 2.6), discussed quantitative methods for evaluating the environmental factors that tend to dominate the final selection process. These methods

generally involved the selection of a set of the important environmental considerations, assigning a weighting factor to each consideration, and then rating each site as objectively as possible for each consideration. The authors generally agreed that the assignment of weighting factors was highly subjective. Fischer observed that the quantitative methods, although useful in the site-selection process, were not a substitute for good judgment and careful evaluation. Several of the authors noted that an important reason for using a quantitative method is to have a permanent record of the rating and weight given each factor in reaching a site-selection decision. Such data may be needed in the licensing process, which takes place months or years after the original site-selection decision has been made.

A standing-room-only audience attended the special 1-hr session at the end of Session 2 to hear Levine's presentation (Paper 2.8) on the Rasmussen report.² Since this draft report has been widely distributed and even more widely reported, the detailed contents will not be repeated here. The study applies only to light-water reactors (LWRs) of current U. S. design. Levine's list of "insights" from the study include the following: (1) the risks are very small, (2) core meltdown is not necessarily a catastrophe, (3) the probability of nuclear accidents affecting the general public is much smaller than that of several types of nonnuclear accidents, (4) the largest consequences of nuclear accidents are much smaller than many have believed, and (5) typical reactor accident sequences are determined by the following five factors: the probability of some initiating event (such as a pipe break), the probability of failure of a single engineered safety system, the probability of occurrence of a particular mode of containment failure, the probability that a particular weather condition will exist, and the probability that a particular population density will be exposed to the radioactivity released.

The heart of the analysis is the construction of an "event diagram" on which all possible paths leading to the release of activity can be traced. A probability of occurrence is assigned to each event along the path, and finally an overall probability of a release by any path can be calculated. Levine emphasized that a significant magnitude of released radioactivity was obtained only via core meltdown and that the principal contributors to overall reactor risk derived from failure to remove decay heat from a shutdown reactor and from core melting associated with small pipe breaks plus some safety-system failure. A "failure to scram," for example, made only a small contribution to the

overall risk. One member of the audience commented that the fault diagrams should be studied by reactor designers to uncover possibilities for (1) significantly reducing the probability of accidents and (2) reducing costs by eliminating engineered safety features that do not significantly affect the accident probability.

Levine noted that, although the study showed that the current power reactors are safe relative to other human endeavors, the reactors are safe not because of the study but because of the painstaking design and careful operation of these devices and that these efforts must be continued if reactors are to remain safe.

When asked how he could be sure that the accident probabilities were really as low as indicated in the study, Levine replied that considerably more operating experience would be required to confirm the predicted probabilities. However, he noted that it was in the nature of accidents that minor accidents occur much more frequently than major accidents and that the incidence of serious core overheating should be an order of magnitude more frequent than the incidence of core meltdown. To date, neither a core meltdown nor a serious core overheat has occurred in commercial LWRs.

SESSION 3: TECHNICAL ISSUES IN SITING

Session 3, except for Woodard's paper on the evaluation of meteorological data (Paper 3.6), was largely devoted to radioactivity release as a factor in the site-selection process. Beattie (Paper 3.1) reported that in the United Kingdom, a linear relation between the probability and magnitude of radioactivity release is assumed in assessing the probable effects of a release at a given site. He concluded that the probable number of casualties or the probable financial loss differs by less than a factor of 10 between remote and semiurban sites and that this order-of-magnitude difference is not large when applied to probabilities on the order of 10^{-6} .

Kohler (Paper 3.2) described a site population factor (SPF) developed for comparing population distributions by weighting each element of population by its distance from a given site. The weighting factor selected was the radial distance to the ± 1.5 power, and the weighted population was then normalized to that of a hypothetical site with a uniform population density of 1000 people per square mile. Kohler emphasized that the SPF was intended only as a tool for comparing potential sites with variously skewed

population distributions. However, the audience indicated considerable interest in whether the SPF might eventually be used as a criterion in siting regulation.

Moffette (Paper 3.5) discussed the potential siting advantage of the high-temperature gas-cooled reactor (HTGR), in that the required exclusion area would generally be smaller than that for a similar LWR. This, he noted, is due to a much slower release of fission products in the design-basis accident, so that the maximum hypothetical fission-product release during the first 2 hr is insignificant in relation to the 10 CFR 100 2-hr guideline for dose at the exclusion-area boundary.

Woodard (Paper 3.6) analyzed a large amount of meteorological data from 10 different reactor sites to determine, among other things, if reasonable site diffusion estimates can be made with less than a full year of data or less than 90% data recovery. His results show that, compared with the use of a full year's data as a reference, the deviation of the diffusion estimate based on 50% data recovery (dropping every other month) did not exceed 15%; based on dropping an entire season did not exceed 18%; and based on only a half-year (spring-summer or fall-winter) did not exceed 40%. In the discussion following the presentation, Woodard observed that, although not necessarily advocating that the AEC relax its requirement for data recovery for regulatory purposes, he felt that considerably less than a year's data at 90% recovery would be adequate as a base for some reactor design decisions.

SESSION 4: SPECIAL SITING ISSUES

This session provided a catchall for several unrelated but interesting ideas. The first three papers were devoted almost entirely to environmental rather than safety considerations. Miner (Paper 4.1) discussed some of the effects of the impending shortage of freshwater in California on the choice of cooling for future power plants. He noted that the makeup requirements for closed-cycle cooling systems for all the plants expected to be built between now and 1990 would double the predicted shortage in freshwater supply for that year. He discussed several alternative sources of makeup water, including recycled waste water and seawater. However, he favored the use of once-through cooling with seawater for ocean sites, noting that studies of existing plants show that their thermal discharges have not created widespread ecological imbalance in the adjoining ocean, even when, in some instances, the

thermal discharges did not meet the temperature standards.

Cushman (Paper 4.2) discussed one specific environmental objection to the creation of cooling lakes, i.e., that they remove agricultural land from production. He noted that the energy saved in pumping for a cooling pond compared with a natural-draft cooling tower amounted to 18 MW for a typical two-unit station. The value of this energy saving is over \$2 million annually, whereas the value of the corn that could be produced on the land is less than \$0.1 million (at 1973 prices). This, he argued, is a reasonable indication of the value of these alternative uses of the land to society. He also noted that modern agriculture is a heavy energy user (e.g., in tractor fuel and the energy required to produce fertilizer). Some of the energy saving from the use of a cooling lake could be applied to increasing production on the remaining agricultural land.

Bonsack (Paper 4.3) discussed potential sites for floating nuclear plants in addition to the proposed offshore sites. Offshore sites have the psychological and aesthetic advantages (over coastal sites, for instance) that the plants are nearly out of sight and the practical advantages of low population density and availability of cooling water. Although these advantages may be lost at other sites, certain technical advantages, such as standardization and the potential cost savings from assembly-line production, would be retained. The other sites considered were onshore or riverine, where the plant would be moored in a prepared basin, and protected ocean sites near the shore, where the plant might be moored in a dredged or natural basin and connected to land by a causeway. The siting considerations for such locations would be similar to those for land-based plants.

Doan (Paper 4.4) presented the results of a review of underground nuclear siting performed for the AEC by United Engineers and Constructors Inc. and Acres American Inc. In this concept the nuclear plant, including the conventional containment system, would be placed in a cavern excavated from competent bedrock. The advantages are an additional barrier to fission-product release; protection from surface hazards; the possibility of flooding the entire cavern in the event of a core meltdown accident, thus providing an emergency heat sink; and the psychological and aesthetic advantages of placing the plant out of sight. It is also possible that an underground plant may be more resistant to seismic damage than a surface structure.

Disadvantages are the possibility of accidental flooding; restricted access, which could hinder inspection and maintenance; and higher cost (direct costs were estimated to be 5 to 20% higher).

A number of hydroelectric and other power facilities have been built underground. However, a full-scale nuclear plant would require a larger cavern than has been constructed heretofore. The construction was judged to be feasible with current technology.

In contrast to Doan's optimistic assessment of underground siting, Véya (Paper 5.7) of Switzerland took a dim view of this concept. In answering questions following his paper on siting of nuclear plants in Switzerland, Véya anticipated new safety problems in connection with geology, hydrology, rock mechanics, and lining stability. He described the underground nuclear plant as a full-scale experiment with high technological risk and voiced the opinion that the construction of large underground caverns should be a well-developed technology before anything so complex as a nuclear power plant was placed inside one.

Hornyik (Paper 4.5) presented an analytical method for assessing the probability that a nuclear plant would be damaged as the result of an accident involving hazardous cargo in transit on a nearby highway or railroad.

SESSION 5: SITING EXPERIENCE

This session included a large variety of utility siting experiences in the United States and Europe. Results of a siting survey developed for the state of Ohio by NUS Corporation (Paper 5.6) probably belonged in Session 2, where methodology of siting was the primary emphasis. The primary themes heard in the papers included public-acceptance problems, the difficulties in making trade-offs in site selection, and the scarcity of perfect sites. Some utilities now conduct fairly explicit siting exercises, whereas others rely largely on implicit criteria, as indicated by the Atomic Industrial Forum survey of siting practices at 26 major utilities (Paper 5.1). Implicit in many questions raised and in several of the papers was the concern that the utilities do not know, prior to site selection, all the criteria that sites must meet. The intensive questioning of AEC's population-site comparison work (Paper 3.2) also demonstrated the concern that new siting requirements will be made retroactive. Both the Swiss siting experience (Paper 5.7) and the French siting plans (Paper 5.8) are particularly interesting because of

different criteria on population density, aesthetics, and transmission lines.

Several papers demonstrated the Atomic Industrial Forum point that few perfect sites remain and that choices invariably involve trade-offs. Leitzell (Paper 5.2) of Commonwealth Edison Company described several less-optimal solutions involving nonnuclear units necessitated by delay of planned nuclear units. For instance, when planned base-load nuclear plants could not be made available in time, smaller, more-expensive oil- and coal-fired peaking or base-load units were built. According to Leitzell, the most desirable locations and/or sequences have been sacrificed because of the delays. Commonwealth Edison's goal remains that of siting four or more large nuclear units per site on geographically dispersed sites. The Public Service Electric & Gas Company of New Jersey (Paper 5.3) now favors floating and out-of-area nuclear plants since no further acceptable sites can be located in its high-population-density service area. Scoville (Paper 5.5) outlined the detailed site-selection process at Douglas Point, involving judgments and trade-offs on population density (near and far), foundation costs, sensitivity of aquatic ecology, and land costs.

Public-acceptance problems have led to changes in several utilities' plans. For example, Commonwealth Edison has reduced the size of cooling lakes at its Braidwood and La Salle stations; the utility is constrained to inland sites as a result of the 5-year moratorium on Lake Michigan siting declared by the Illinois Pollution Control Board. In response to citizen complaints of lack of public access to the siting-selection process, Southern California Edison has initiated a community participation plan for its proposed Eastern Mojave Desert site. The 2-year-old California Coastal Zone Conservation Commission initially denied Southern California Edison a permit for expansion of its San Onofre plant. After court action, the California Coastal Zone Conservation Commission approved the project with an additional list of substantive provisos. It is apparent that changing public and (hence) regulatory expectations are sources of considerable puzzlement and frustration to most utilities. Power producers view such changes primarily in terms of additional costs and delays in implementation of their plans. A few utilities are now taking initial steps to enable themselves to predict, monitor, and/or include public expectations and input so that costly and painful belated planning turnarounds can be minimized.

The unhappy experience of the Public Service Electric & Gas Company of New Jersey in the belated

disapproval by AEC of its Newbold Island site was discussed guardedly by the utility. Utilities' frustration with changing AEC guidelines was apparent also in the questions.

The Swiss experience in moving rapidly into nuclear power generation (Paper 5.7) since 1965 (all hydropower before) already demonstrates a marked siting difference from U. S. practice. Lightly populated Alpine sites are disqualified because these areas are viewed as being too important for aesthetic and recreational purposes. Switzerland's high population density is seen as requiring placement of existing and future electricity-generating stations close to or in urban load centers. Véya observed that this parameter changes the focus *from siting suitability to safety adequacy of specific available sites* [italicized by the Editor for emphasis] since all are located near load centers. Population density at Swiss as well as other European nuclear sites is well above that of the most populated U. S. reactor sites, such as Indian Point or Zion. Because right-of-way clearance for transmission lines is a major stumbling block to site approval, new siting guidelines are expected to include a statement on "no new overhead lines." The evolution of environmental issues for Swiss citizens led to a 6-year delay between the licensing of the third and fourth nuclear power reactors in Switzerland and involved the switch from freshwater once-through cooling to cooling towers. Swiss electric-power producers now accept the idea that "the environmental issue invalidated all prior economic arguments such as site development or plant costs." Although Véya made no comparisons, it seems doubtful that environmental issues have achieved such primacy in U. S. power-plant siting decision making, despite greatly increased attention from both utilities and government.

Other interesting differences in Swiss experience include (1) site permits given for two to three reactor types, allowing considerably more freedom in technical and economic decision making; (2) utilities' confidence in safety standards despite lack of written requirements; and (3) utilities' practice of awarding equipment contracts only after having site approval in hand. Presumably all European countries face the Swiss problem of international considerations when siting plants near their borders. Active opposition from citizens of three neighboring countries has arisen and continues from the Austrian province of Vorarlberg; France and West Germany were successful in soothing complaints from their citizens about bordering Swiss nuclear plants.

French constraints in nuclear power-plant siting include a preponderance of low-flow rivers and a highly dispersed rural population. These factors prevent use of cooling lakes and favor coastal siting, according to Bellin and Toureau (Paper 5.8).

SESSION 6: SITING AND SOCIAL ISSUES

Despite the fact that Session 6 was the final conference session and the only one with programmed competition (the Trojan plant tour), about 100 persons attended. Public acceptance of nuclear power-plant siting was the principal focus—its origin, cultivation, parameters, and measurement.

The known social impacts of nuclear power plants, problems in social-impact assessment, possibilities for greater public participation in siting decisions, and tentative siting guidelines were reviewed by Peelle (Paper 6.3). The public interest in nuclear power plants, which has increased with growth of the industry, with changing public evaluation of the natural environment, and with reevaluation of the growth ethic, is a reaction in part to AEC's and utilities' procedures, policies, and attitudes of the past. Peelle warned that the basis of our siting problems is much broader and more well-founded than is the basis of some other problems in industry and stated that the AEC likes to admit and that "public concern is not merely the result of a few obstreperous troublemakers stirring up protest." Preconditions for public acceptance include greater public involvement in siting decisions and mitigation of impacts from plant construction and operation. Peelle further stated that new approaches to public acceptance are needed (that the old ones are not working) and urged devising new mechanisms for public input into siting decisions, emphasizing the need for two-way communication. Most of the audience comment in this session was directed to this paper.

Three papers dealt with methods for determining public acceptance or incorporating community judgments into power-plant siting decisions.

How to combine societal and technical judgments in environmental decision making was the thorny problem addressed by Burnham et al. (Paper 6.1). Community judgments on the relative importance of eight environmental-societal criteria were measured using three invited participant groups. Citizens' opinions on the acceptability of various siting and design options were then sought and weighted according to

the rankings previously established. The highest ranking criteria (all 0.8 or 0.7) were land use, water impacts, health and safety, and ecological impact. Burnham illustrated how such weighted rankings could result in different conclusions concerning specific sites than when only technical and economic factors were considered.

The undoubted visual highlight of the session was the presentation by Jones (Paper 6.2) of a method for quantification of aesthetic values in environmental decision making. Applying the principles given in Paper 6.1, Jones described assumptions and analytical techniques used to assess visual impact of various proposed plant designs on three selected views at the Rancho Seco and Trojan sites. The assessment technique holds good promise of providing a basis for judgment, not only for landscape architects but also for plant designers and site selectors.

Sucov (Paper 6.4) described a valuable methodological advance based on a case study using conflict theory as an approach to assessing community acceptance. This pragmatic and careful work attempts to measure and assess operating strengths of various community groups. Although this method probably cannot be applied before siting plans have been announced and issues have been developed, it does provide a means of assessing the relative strengths of community forces (occupational and environmental groups, influential individuals, and the general public) after the siting process is well advanced. According to Sucov, work is now in progress on another technique to permit earlier assessment.

Leaving methodology, Stimac's case study of Puget Sound Power & Light's experience with zoning arrangements for its Skagit plant (Paper 6.5) again raised larger social concerns over siting. Conclusions suggested by the many interesting points in his account include:

1. Public approval and acceptance of nuclear power plants are becoming more sophisticated, at least in Washington State. Even in a climate of substantial local approval, spot zoning was not acceptable and resulted in a 1-year delay of (unilateral) utility plans while other arrangements were worked out.

2. The placement of industrial zones in rural areas presents several new problems and supports the need for predesignated sites (see Paper 1.1).

3. The contract zoning solution devised by Puget Sound Power & Light Company is a creative and highly significant innovation in siting because it (1) provides significant and binding local input for the siting decision process, (2) mitigates expected social and community impacts by closing some major "disassociation gaps" on cost and benefit allocation within society, and (3) protects legitimate interests of both parties by contractual arrangement.

CONCLUSION

The conference was unique among ANS meetings in that most papers recognized, at least implicitly, the growing importance of social impacts of power-plant siting. Previous siting conferences have concentrated almost exclusively on quantifiable engineering aspects, such as seismic or thermohydraulic effects. Few non-engineers or nonphysical scientists attended except for a half-dozen persons giving papers. Perhaps more interchange between ANS members and lawyers, social scientists, and public officials can be built into future conferences. Highly significant decisions affecting the future of nuclear power will be made in the next 5 years without benefit of such interchange unless a more active role is taken in encouraging it. In the opinion of one author (Peelle), the majority of these decisions will involve nontechnical people considering nontechnical issues and constraints. The nontechnologist input to the Portland meeting on siting was a good start, but we need more of the same on other issues and at many more levels.

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APPENDICES

APPENDIX A

REACTOR SITE CRITERIA

[Code of Federal Regulations - Energy, Title 10
Part 100 (Apr. 30, 1975)]

Sec.
100.1 Purpose.
100.2 Scope.
100.3 Definitions.

SITE EVALUATION FACTORS

100.10 Factors to be considered when evaluating sites.
100.11 Determination of exclusion area, low population zone, and population center distance.

APPENDIX A

SCIENTIFIC AND GEOLOGIC CRITERIA FOR
NUCLEAR POWER PLANTS

AUTHORITY: Sec. 202, Pub. L. 93-438, 88
Stat. 1244 (42 U.S.C. 5842)

§ 100.1 Purpose.

(a) It is the purpose of this part to describe criteria which guide the Commission in its evaluation of the suitability of proposed sites for stationary power and testing reactors subject to Part 50 of this chapter.

(b) Insufficient experience has been accumulated to permit the writing of detailed standards that would provide a quantitative correlation of all factors significant to the question of acceptability of reactor sites. This part is intended as an interim guide to identify a number of factors considered by the Commission in the evaluation of reactor sites and the general criteria used at this time as guides in approving or disapproving proposed sites. Any applicant who believes that factors other than those set forth in the guide should be considered by the Commission will be expected to demonstrate the applicability and significance of such factors.

§ 100.2 Scope.

(a) This part applies to applications filed under Part 50 of this chapter for stationary power and testing reactors.

(b) The site criteria contained in this part apply primarily to reactors of a general type and design on which experience has been developed, but can also be applied to other reactor types. In particular, for reactors that are novel in design and unproven as prototypes or pilot plants, it is expected that these basic criteria will be applied in a manner that takes into account the lack

of experience. In the application of these criteria which are deliberately flexible, the safeguards provided—either site isolation or engineered features—should reflect the lack of certainty that only experience can provide.

§ 100.3 Definitions.

As used in this part:

(a) "Exclusion area" means that area surrounding the reactor, in which the reactor licensee has the authority to determine all activities including exclusion or removal of personnel and property from the area. This area may be traversed by a highway, railroad, or waterway, provided these are not so close to the facility as to interfere with normal operations of the facility and provided appropriate and effective arrangements are made to control traffic on the highway, railroad, or waterway, in case of emergency, to protect the public health and safety. Residents within the exclusion area shall normally be prohibited. In any event, residents shall be subject to ready removal in case of necessity. Activities unrelated to operation of the reactor may be permitted in an exclusion area under appropriate limitations, provided that no significant hazards to the public health and safety will result.

(b) "Low population zone" means the area immediately surrounding the exclusion area which contains residents, the total number and density of which are such that there is a reasonable probability that appropriate protective measures could be taken in their behalf in the event of a serious accident. These guides do not specify a permissible population density or total population within this zone because the situation may vary from case to case. Whether a specific number of people can, for example, be evacuated from a specific area, or instructed to take shelter, on a timely basis will depend on many factors such as location, number and size of highways, scope and extent of advance 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 densely populated center containing more than about 25,000 residents.

(d) "Power reactor" means a nuclear reactor of a type described in § 50.31(b)

or 50.32 of this chapter designed to produce electrical or heat energy.

(e) "Testing reactor" means a "testing facility" as defined in § 50.3 of this chapter.

SITE EVALUATION FACTORS

§ 100.10 Factors to be considered when evaluating sites.

Factors considered in the evaluation of sites include those relating both to the proposed reactor design and the characteristics peculiar to the site. It is expected that reactors will reflect through their design, construction and operation an extremely low probability for accidents that could result in release of significant quantities of radioactive fission products. In addition, the site location and the engineered features included as safeguards against the hazardous consequences of an accident, should one occur, should insure a low risk of public exposure. In particular, the Commission will take the following factors into consideration in determining the acceptability of a site for a power or testing reactor:

(a) Characteristics of reactor design and proposed operation including:

(1) Extended use of the reactor including the proposed maximum power level and the nature and inventory of contained radioactive materials;

(2) The extent to which generally accepted engineering standards are applied to the design of the reactor;

(3) The extent to which the reactor incorporates unique or unusual features having a significant bearing on the probability or consequences of accidental release of radioactive materials;

(4) The safety features that are to be engineered into the facility and those barriers that must be breached as a result of an accident before a release of radioactive material to the environment can occur.

(b) Population density and the characteristics of the site environment, including the exclusion area, low population zone, and population center distance.

(c) Physical characteristics of the site, including seismology, meteorology, geology and hydrology.

(1) Appendix A, "Scientific and Geologic Site Criteria for Nuclear Power Plants," describes the nature of investigations re-

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quired to obtain the geologic and seismic data necessary to determine site suitability; and to provide reasonable assurance that a nuclear power plant can be constructed and operated at a proposed site without undue risk to the health and safety of the public. It describes procedures for determining the quantitative vibratory ground motion design basis at a site due to earthquakes and describes information needed to determine whether and to what extent a nuclear power plant need be designed to withstand the effects of surface faulting.

(2) Meteorological conditions at the site and in the surrounding area should be considered.

(3) Geological and hydrological characteristics of the proposed site may have a bearing on the consequences of an escape of radioactive material from the facility. Special precautions should be planned if a reactor is to be located at a site where a significant quantity of radioactive effluent might accidentally flow into nearby streams or rivers or might find ready access to underground water tables.

(d) Where unfavorable physical characteristics of the site exist, the proposed site may nevertheless be found to be acceptable if the design of the facility includes appropriate and adequate compensating engineering safeguards.

§ 100.11 Determination of exclusion area, low population zone, and population center distance.

(a) As an aid in evaluating a proposed site, an applicant should assume a fission product release¹ from the core, the expected demonstrable leak rate from the containment and the meteorological conditions pertinent to his site to derive an exclusion area, a low population zone and population center distance. For the purpose of this analysis, which shall set forth the basis for the numerical values used, the applicant should determine the following:

¹ The fission product release assumed for these calculations should be based upon a major accident, hypothesized for purposes of site analysis or postulated from considerations of possible accidental events, that would result in potential hazards not exceeded by those from any accident considered credible. Such accidents have generally been assumed to result in substantial meltdown of the core with subsequent release of appreciable quantities of fission products.

² The whole body dose of 25 rem referred to above corresponds numerically to the dose in a lifetime accidental or emergency dose for radiation workers which, according to NCRP recommendations may be regarded in the determination of their radiation exposure rates (see NCRP Handbook 60 dated July 3, 1960). However, neither its use nor that of the 300 rem value for thyroid exposure as set forth in these site criteria guides are intended to imply that these numbers constitute acceptable limits for emergency doses to the public under accident conditions. Rather, the 25 rem whole body value and the 300 rem thyroid value have been set forth in these guides as reference values, which can be used in the evaluation of reactor sites with respect to potential reactor accidents of exceedingly low probability of occurrence, and low risk of public exposure to radiation.

(1) An exclusion area of such size that an individual located at any point on its boundary for two hours immediately following onset of the postulated fission product release would not receive a total radiation dose to the whole body in excess of 25 rem² or a total radiation dose in excess of 300 rem² to the thyroid from iodine exposure.

(2) A low population zone of such size that an individual located at any point on its outer boundary who is exposed to the radioactive cloud resulting from the postulated fission product release (during the entire period of its passage) would not receive a total radiation dose to the whole body in excess of 25 rem or a total radiation dose in excess of 300 rem to the thyroid from iodine exposure.

(3) A population center distance of at least one and one-third times the distance from the reactor to the outer boundary of the low population zone. In applying this guide, due consideration should be given to the population distribution within the population center.

Where very large cities are involved, a greater distance may be necessary because of total integrated population dose consideration.

(b) For sites for multiple reactor facilities consideration should be given to the following:

(1) If the reactors are independent to the extent that an accident in one reactor would not initiate an accident in another, the size of the exclusion area, low population zone and population center distance shall be fulfilled with respect to each reactor individually. The envelopes of the plan overlay of the areas so calculated shall then be taken as their respective boundaries.

(2) If the reactors are interconnected to the extent that an accident in one reactor could affect the safety of operation of any other, the size of the exclusion area, low population zone and population center distance shall be based upon the assumption that all interconnected reactors emit their postulated fission product releases simultaneously. This requirement may be reduced in relation to the degree of coupling between reactors, the probability of concomitant accidents and the probability that an individual would not be exposed to the radiation effects from simultaneous releases. The applicant would be expected to justify to the satisfaction of the Commission the basis for such a reduction in the source term.

(3) The applicant is expected to show that the simultaneous operation of multiple reactors at a site will not result in total radioactive effluent releases beyond the allowable limits of applicable regulations.

Note: For further guidance in developing the exclusion area, the low population zone, and the population center distance, reference is made to Technical Information Document 14844, dated March 23, 1962, which contains a procedural method and a sample calculation that result in distances roughly reflecting current siting practices of the Commission. The calculations described in Technical Information Document 14844 may be used as a point of departure for consideration of particular site requirements which may result from evaluation of the characteristics of a particular reactor, its purpose and method of operation.

Copies of Technical Information Document 14844 may be obtained from the Commission's Public Document Room, 1717 H Street NW, Washington, D.C. or by writing the Director of Nuclear Reactor Regulation, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555.

APPENDIX A.—SEISMIC AND GEOLOGIC SETTING CRITERIA FOR NUCLEAR POWER PLANTS

I. PURPOSE

General Design Criterion 2 of Appendix A to Part 50 of this chapter requires that nuclear power plant structures, systems, and components important to safety be designed to withstand the effects of natural phenomena such as earthquakes, tornadoes, hurricanes, floods, tsunamis, and wildfires without loss of capability to perform their safety functions. It is the purpose of these criteria to set forth the principal seismic and geologic considerations which guide the Commission in its evaluation of the suitability of proposed sites for nuclear power plants and the suitability of the plant design based established in consideration of the seismic and geologic characteristics of the proposed sites.

These criteria are based on the limited geophysical and geological information available to date concerning faults and earthquake occurrence and effect. They will be revised as necessary when more complete information becomes available.

II. SCOPE

These criteria, which apply to nuclear power plants, describe the nature of the investigations required to obtain the geologic and seismic data necessary to determine site suitability and provide reasonable assurance that a nuclear power plant can be constructed and operated at a proposed site without undue risk to the health and safety of the public. They describe procedures for determining the quantitative vibratory ground motion design basis at a site due to earthquakes and describe information needed to determine whether and to what extent a nuclear power plant need be designed to withstand the effects of surface faulting. Other geologic and seismic factors required to be taken into account in the siting and design of nuclear power plants are identified.

The investigations described in this appendix are within the scope of investigations permitted by § 50.10(c)(1) of this chapter.

Each applicant for a construction permit shall investigate all seismic and geologic factors that may affect the design and operation of the proposed nuclear power plant irrespective of whether such factors are explicitly included in these criteria. Additional investigations and/or more conservative determinations than those included in these criteria may be required for sites located in areas having complex geology or in areas of high seismicity. If an applicant believes that the particular seismology and geology of a site indicate that some of these criteria, or portions thereof, need not be satisfied, the specific sections of these criteria should be identified in the license application, and supporting data to justify clearly such departures should be presented.

These criteria do not address investigations of volcanic phenomena required for sites located in areas of volcanic activity. Investigations of the volcanic aspects of such sites will be determined on a case-by-case basis.

III. DEFINITIONS

As used in these criteria:

- (a) The "magnitude" of an earthquake is a measure of the size of an earthquake and is related to the energy released in the form of seismic waves. "Magnitude" means the numerical value on a Richter scale.
- (b) The "intensity" of an earthquake

a measure of its effects on man, on man-built structures, and on the earth's surface at a particular location. "Intensity" means the numerical value on the Modified Mercalli scale.

(c) The "Safe Shutdown Earthquake" is that earthquake which is based upon an evaluation of the maximum earthquake potential considering the regional and local geology and seismology and specific characteristics of local subsurface material. It is that earthquake which produces the maximum vibratory ground motion for which certain structures, systems, and components are designed to remain functional. These structures, systems, and components are those necessary to ensure:

(1) The integrity of the reactor coolant pressure boundary.

(2) The capability to shut down the reactor and maintain it in a safe shutdown condition, or

(3) The capability to prevent or mitigate the consequences of accidents which could result in potential offsite exposures comparable to the guideline exposures of this part.

(d) The "Operating Basis Earthquake" is that earthquake which, considering the regional and local geology and seismology and specific characteristics of local subsurface material, could reasonably be expected to affect the plant site during the operating life of the plant; it is that earthquake which produces the vibratory growth motion for which those features of the nuclear power plant necessary for continued operation without undue risk to the health and safety of the public are designed to remain functional.

(e) A "fault" is a tectonic structure along which differential slippage of the adjacent earth materials has occurred parallel to the fracture plane. It is distinct from other types of ground disruptions such as landslides, fissures, and craters. A fault may have gouge or breccia between its two walls and includes any associated monoclinical flexure or other similar geologic structural feature.

(f) "Surface faulting" is differential ground displacement at or near the surface caused directly by fault movement and is distinct from nontectonic types of ground disruptions, such as landslides, fissures, and craters.

(g) A "capable fault" is a fault which has exhibited one or more of the following characteristics:

(1) Movement at or near the ground surface at least once within the past 25,000 years or movement of a recurring nature within the past 500,000 years.

(2) Macro-seismicity instrumentally determined with records of sufficient precision to demonstrate a direct relationship with the fault.

(3) A structural relationship to a capable fault according to characteristics (1) or (2) of this paragraph such that movement on one could be reasonably expected to be accompanied by movement on the other.

In some cases, the geologic evidence of past activity at or near the ground surface along a particular fault may be obscured at a particular site. This might occur, for example, at a site having a deep overburden. For these cases, evidence may exist elsewhere along the fault from which an evaluation of its characteristics in the vicinity of the site can be reasonably based. Such evidence shall be used in determining whether the fault is a capable fault within this definition.

Notwithstanding the foregoing paragraphs III(g)(1), (2) and (3), structural association of a fault with geologic structural features which are geologically old (at least pre-Quaternary) such as many of those found in the Eastern region of the United

States shall, in the absence of conflicting evidence, demonstrate that the fault is not a capable fault within this definition.

(h) A "tectonic province" is a region of the North American continent characterized by a relative consistency of the geologic structural features contained therein.

(i) A "tectonic structure" is a large scale dislocation or distortion within the earth's crust. Its extent is measured in miles.

(j) A "zone requiring detailed faulting investigation" is a zone within which a nuclear power reactor may not be located unless a detailed investigation of the regional and local geologic and seismic characteristics of the site demonstrates that the need to design for surface faulting has been properly determined.

(k) The "control width" of a fault is the maximum width of the zone containing mapped fault traces, including all faults that can be reasonably inferred to have occurred differential movement during Quaternary times and which join or can reasonably be inferred to join the main fault trace, measured within 10 miles along the fault's trend in both directions from the point of nearest approach to the site. (See Figure 1 of this appendix.)

(l) A "response spectrum" is a plot of the maximum responses (acceleration, velocity or displacement) of a family of idealized single-degrees-of-freedom damped oscillators against natural frequencies (or periods) of the oscillators to a specified vibratory motion input at their supports.

IV. REQUIRED INVESTIGATIONS

The geologic, seismic and engineering characteristics of a site and its environs shall be investigated in sufficient scope and detail to provide reasonable assurance that they are sufficiently well understood to permit an adequate evaluation of the proposed site, and to provide sufficient information to support the determinations required by these criteria and to permit adequate engineering solutions to actual or potential geologic and seismic effects at the proposed site. The size of the region to be investigated and the type of data pertinent to the investigations shall be determined by the nature of the region surrounding the proposed site. The investigations shall be carried out by a review of the pertinent literature and field investigations and shall include the steps outlined in paragraphs (a) through (c) of this section.

(a) *Required Investigation for Vibratory Ground Motion.* The purpose of the investigations required by this paragraph is to obtain information needed to describe the vibratory ground motion produced by the Safe Shutdown Earthquake. All of the steps in paragraphs (a)(5) through (a)(8) of this section need not be carried out if the Safe Shutdown Earthquake can be clearly established by investigations and determinations of a lesser scope. The investigations required by this paragraph provide an adequate basis for selection of an Operating Basis Earthquake. The investigations shall include the following:

(1) Determination of the lithologic, stratigraphic, hydrologic, and structural geologic conditions of the site and the region surrounding the site, including its geologic history;

(2) Identification and evaluation of tectonic structures underlying the site and the region surrounding the site, whether buried or expressed at the surface. The evaluation should consider the possible effects caused by man's activities such as withdrawal of fluid from or addition of fluid to the subsurface, extraction of minerals, or the loading effects of dams or reservoirs;

(3) Evaluation of physical evidence concerning the behavior during prior earthquakes of the surficial geologic materials and the substrata underlying the site from the lithologic, stratigraphic, and structural geologic

The "Safe Shutdown Earthquake" defines that earthquake which has commonly been referred to as the "Design Basis Earthquake."

logic studies

(4) Determination of the static and dynamic engineering properties of the materials underlying the site. Included should be properties needed to determine the behavior of the underlying material during earthquakes and the characteristics of the underlying material in transmitting earthquake-induced motions to the foundations of the plant, such as seismic wave velocities, density, water content, porosity, and strength.

(5) Listing of all historically reported earthquakes which have affected or which could reasonably be expected to have affected the site, including the date of occurrence and the following measured or estimated data: magnitude or highest intensity, and a plot of the epicenter or location of highest intensity. Where historically reported earthquakes could have caused a maximum ground acceleration at least one-third the acceleration of gravity (0.1g) at the foundations of the proposed nuclear power plant structures, the acceleration or intensity and duration of ground shaking at these foundations shall also be estimated. Since earthquakes have been reported in terms of various parameters such as magnitude, intensity at a given location, and effect on ground, structure, and people at a specific location, some of these data may have to be estimated by use of appropriate empirical relationships. The comparative characteristics of the material underlying the epicentral location or region of highest intensity and of the material underlying the site in transmitting earthquake vibratory motion shall be considered.

(6) Correlation of epicenters or locations of highest intensity of historically reported earthquakes, where possible, with tectonic structures any part of which is located within 200 miles of the site. Epicenters or locations of highest intensity which cannot be reasonably correlated with tectonic structures shall be identified with tectonic provinces any part of which is located within 200 miles of the site.

(7) For faults, any part of which is within 200 miles of the site and which may be of significance in establishing the Safe Shutdown Earthquake, determination of whether these faults are to be considered as capable faults.* This determination is required in order to permit appropriate consideration of the geologic history of such faults in establishing the Safe Shutdown Earthquake. For guidance in determining which faults may be of significance in determining the Safe Shutdown Earthquake, Table 1 of this appendix presents the minimum length of faults to be considered versus distance from site. Capable faults of lesser length than those indicated in Table 1 and faults which are not capable faults need not be considered in determining the Safe Shutdown Earthquake, except where unusual circumstances indicate such consideration is appropriate.

*If the Safe Shutdown Earthquake can be associated with a fault closer than 200 miles to the site, the procedures of paragraphs (a) (7) and (a) (8) of this section need not be carried out for successively more remote faults.

In the absence of absolute dating, evidence of recency of movement may be obtained by applying relative dating technique to ruptured, offset, warped or otherwise structurally disturbed surface or near surface materials or geomorphic features.

The applicant shall evaluate whether or not a fault is a capable fault with respect to the characteristics outlined in paragraphs III(g)(1), (2), and (3) by conducting a reasonable investigation using suitable geologic and geophysical techniques.

TABLE 1

Distance from the site (miles):	Minimum length:
0 to 20.....	1
Greater than 20 to 50.....	5
Greater than 50 to 100.....	10
Greater than 100 to 150.....	20
Greater than 150 to 200.....	40

(8) For capable faults, any part of which is within 200 miles of the site and which may be of significance in establishing the Safe Shutdown Earthquake, determination of:

- (i) The length of the fault;
- (ii) The relationship of the fault to regional tectonic structures; and
- (iii) The nature, amount, and geologic history of displacements along the fault, including particularly the estimated amount of the maximum Quaternary displacement related to any one earthquake along the

(b) *Required Investigation for Surface Faulting.* The purpose of the investigations required by this paragraph is to obtain information to determine whether and to what extent the nuclear power plant need be designed for surface faulting. If the design basis for surface faulting can be clearly established by investigations of a lesser scope, not all of the steps in paragraphs (b)(4) through (b)(7) of this section need be carried out. The investigations shall include the following:

- (1) Determination of the lithologic, stratigraphic, hydrologic, and structural geologic conditions of the site and the area surrounding the site, including its geologic history;
- (2) Evaluation of tectonic structures underlying the site, whether buried or expressed at the surface, with regard to their potential for causing surface displacement at or near the site. The evaluation shall consider the possible effects caused by man's activities such as withdrawal of fluid from or addition of fluid to the subsurface, extraction of minerals, or the loading effects of dams or reservoirs;
- (3) Determination of geologic evidence of fault offset at or near the ground surface at or near the site;
- (4) For faults greater than 1000 feet long, any part of which is within 5 miles of the site, determination of whether these faults are to be considered as capable faults;†
- (5) Listing of all historically reported earthquakes which can reasonably be associated with capable faults greater than 1000 feet long, any part of which is within 5 miles of the site, including the date of occurrence and the following measured or estimated data: magnitude or highest intensity, and a plot of the epicenter or region of highest intensity.

†Minimum length of fault (miles) which shall be considered in establishing Safe Shutdown Earthquake.

‡If the design basis for surface faulting can be determined from a fault closer than 5 miles to the site, the procedures of paragraphs (b)(4) through (b)(7) of this section need not be carried out for successively more remote faults.

In the absence of absolute dating, evidence of recency of movement may be obtained by applying relative dating techniques to ruptured, offset, warped or otherwise structurally disturbed surface or near-surface materials or geomorphic features.

The applicant shall evaluate whether or not a fault is a capable fault with respect to the characteristics outlined in paragraphs III(g)(1), (2), and (3) by conducting a reasonable investigation using suitable geologic and geophysical techniques.

(6) Correlation of epicenters or locations of highest intensity of historically reported earthquakes with capable faults greater than 1000 feet long, any part of which is located within 5 miles of the site;

(7) For capable faults greater than 1000 feet long, any part of which is within 5 miles of the site, determination of:

- (i) The length of the fault;
- (ii) The relationship of the fault to regional tectonic structures;
- (iii) The nature, amount, and geologic history of displacements along the fault, including particularly the estimated amount of the maximum Quaternary displacement related to any one earthquake along the fault; and
- (iv) The outer limits of the fault established by mapping Quaternary fault traces for 10 miles along its trend in both directions from the point of its nearest approach to the site.

(c) *Required Investigation for Seismically Induced Floods and Water Waves.* (1) For coastal sites, the investigations shall include the determination of:

- (i) Information regarding distant and locally generated waves or tsunami which have affected or could have affected the site. Available evidence regarding the runup and drawdown associated with historic tsunami in the same coastal region as the site shall also be included;
- (ii) Local features of coastal topography which might tend to modify tsunami runup or drawdown. Appropriate available evidence regarding historic local modifications in tsunami runup or drawdown at coastal locations having topography similar to that of the site shall also be obtained; and
- (iii) Appropriate geologic and seismic evidence to provide information for establishing the design basis for seismically induced floods or water waves from a local offshore earthquake, from local offshore effects of an onshore earthquake, or from coastal subsidence. This evidence shall be determined, to the extent practical, by a procedure similar to that required in paragraphs (a) and (b) of this section. The probable slip characteristics of offshore faults shall also be considered as well as the potential for offshore slides in submarine material.

(2) For sites located near lakes and rivers, investigations similar to those required in paragraph (c)(1) of this section shall be carried out, as appropriate, to determine the potential for the nuclear power plant to be exposed to seismically induced floods and water waves as, for example, from the failure during an earthquake of an upstream dam or from slides of earth or debris into a nearby lake.

V SEISMIC AND GEOLOGIC DESIGN BASIS

(a) *Determination of Design Basis for Vibratory Ground Motion.* The design of each nuclear power plant shall take into account the potential effects of vibratory ground motion caused by earthquakes. The design basis for the maximum vibratory ground motion and the expected vibratory ground motion should be determined through evaluation of the seismology, geology, and the seismic and geologic history of the site and the surrounding region. The most severe earthquakes associated with tectonic structures or tectonic provinces in the region surrounding the site should be identified, considering those historically reported earthquakes that can be associated with these structures or provinces and other relevant factors. If faults in the region surrounding the site are capable faults, the most severe earthquakes associated with these faults should be determined by also considering their geologic history. The vibratory ground motion at the site should be then determined by assuming that the epicenters or locations of highest intensity of the earthquakes are situated at the point on the tectonic structures or tectonic provinces nearest to the site. The earthquakes

which could cause the maximum vibratory ground motion at the site should be designated the Safe Shutdown Earthquake. The specific procedures for determining the design basis for vibratory ground motion are given in the following paragraphs.

(1) *Determination of Safe Shutdown Earthquake.* The Safe Shutdown Earthquake shall be identified through evaluation of seismic and geologic information developed pursuant to the requirements of paragraph IV(a), as follows:

(i) The historic earthquakes of greatest magnitude or intensity which have been correlated with tectonic structures pursuant to the requirements of paragraph (a)(6) of Section IV shall be determined. In addition, for capable faults, the information required by paragraph (a)(8) of Section IV shall also be taken into account in determining the earthquakes of greatest magnitude related to the faults. The magnitude or intensity of earthquakes based on geologic evidence may be larger than that of the maximum earthquakes historically recorded. The accelerations at the site shall be determined assuming that the epicenters of the earthquakes of greatest magnitude or the locations of highest intensity related to the tectonic structures are situated at the point on the structures closest to the site;

(ii) Where epicenters or locations of highest intensity of historically reported earthquakes cannot be reasonably related to tectonic structures but are identified pursuant to the requirements of paragraph (a)(6) of Section IV with tectonic provinces in which the site is located, the accelerations at the site shall be determined assuming that these earthquakes occur at the site.

(iii) Where epicenters or locations of the highest intensity of historically reported earthquakes cannot be reasonably related to tectonic structures but are identified pursuant to the requirements of paragraph (a)(6) of Section IV with tectonic provinces in which the site is not located, the accelerations at the site shall be determined assuming that the epicenters or locations of highest intensity of these earthquakes are at the closest point to the site on the boundary of the tectonic province;

(iv) The earthquake producing the maximum vibratory acceleration at the site, as determined from paragraph (a)(i) through (iii) of this section shall be designated the Safe Shutdown Earthquake for vibratory ground motion, except as noted in paragraph (a)(i)(v) of this section. The characteristics of the Safe Shutdown Earthquake shall be derived from more than one earthquake determined from paragraph (a)(i)(i) through (iii) of this section, where necessary to ensure that the maximum vibratory acceleration at the site throughout the frequency range of interest is included. In the case where a causative fault is near the site, the effect of proximity of an earthquake on the spectral characteristics of the Safe Shutdown Earthquake shall be taken into account. In order to compensate for the limited data, the procedure in paragraphs (a)(i)(i) through (iii) of this section shall be applied in a conservative manner. The maximum vibratory accelerations of the Safe Shutdown Earthquake at each of the various foundation locations of the nuclear power plant structures at a given site shall be determined taking into account the characteristics of the underlying soil material in transmitting the earthquake-induced motions, obtained pursuant to paragraphs (a)(1), (3), and (4) of section IV. The Safe Shutdown Earthquake shall be defined by response spectra corresponding to the maximum vibratory accelerations as outlined in paragraph (a) of section VI; and

(v) Where the maximum vibratory accelerations of the Safe Shutdown Earthquake at the foundations of the nuclear power plant structures are determined to be less

than one-tenth the acceleration of gravity (0.1 g) as a result of the steps required in paragraphs (a)(i)(i) through (iv) of this section, it shall be assumed that the maximum vibratory accelerations of the Safe Shutdown Earthquake at these foundations are at least 0.1 g.

(2) *Determination of Operating Basis Earthquake.* The Operating Basis Earthquake shall be specified by the applicant after considering the seismology and geology of the region surrounding the site. If vibratory ground motion exceeding that of the Operating Basis Earthquake occurs, shutdown of the nuclear power plant will be required. Prior to resuming operations, the licensee will be required to demonstrate to the Commission that no functional damage has occurred to those features necessary for continued operation without undue risk to the health and safety of the public. The maximum vibratory ground acceleration of the Operating Basis Earthquake shall be at least one-half the maximum vibratory ground acceleration of the Safe Shutdown Earthquake.

(b) *Determination of Need to Design for Surface Faulting.* In order to determine whether a nuclear power plant is required to be designed to withstand the effects of surface faulting, the location of the nuclear power plant with respect to capable faults shall be considered. The area over which each of these faults has caused surface faulting in the past is identified by mapping its fault traces in the vicinity of the site. The fault traces are mapped along the trend of the fault for 10 miles in both directions from the point of its nearest approach to the nuclear power plant because, for example, traces may be obscured along portions of the fault. The maximum width of the mapped fault traces, called the control width, is then determined from this map. Because surface faulting has sometimes occurred beyond the limit of mapped fault traces or where fault traces have not been previously recognized, the control width of the fault is increased by a factor which is dependent upon the largest potential earthquake related to the fault.

This larger width delineates a zone, called the zone requiring detailed faulting investigation, in which the possibility of surface faulting is to be determined. The following paragraphs outline the specific procedures for determining the zone requiring detailed faulting investigation for a capable fault.

(1) *Determination of Zone Requiring Detailed Faulting Investigation.* The zone requiring detailed faulting investigation for a capable fault which was investigated pursuant to the requirement of paragraph (b)(7) of Section IV shall be determined through use of the following table:

TABLE 2

Determination of Zone Requiring Detailed Faulting Investigation

Magnitude of earthquake:	Width of zone requiring detailed faulting investigation (See fig. 1)
Less than 5.5.....	1 x control width
5.5-6.4.....	2 x control width
6.5-7.5.....	3 x control width
Greater than 7.5.....	4 x control width

The largest magnitude earthquake related to the fault shall be used in Table 2. This earthquake shall be determined from the information developed pursuant to the requirements of paragraph (b)(i) of Section IV for the fault, taking into account the information required by paragraph (b)(7) of Section IV. The control width used in Table 2 is determined by mapping the outer limits of the fault traces from information developed pursuant to paragraph (b)(7)(iv) of section IV. The control width shall be used in Table 2 unless the characteristics of the fault are obscured for a significant portion of the 10 miles on either side of the point of

nearest approach to the nuclear power plant. In this event, the use in Table 2 of the width of mapped fault traces more than 10 miles from the point of nearest approach to the nuclear power plant may be appropriate.

The zone requiring detailed faulting investigation, as determined from Table 2, shall be used for the fault except where:

(i) The zone requiring detailed faulting investigation from Table 2 is less than one-half mile in width. In this case the zone shall be at least one-half mile in width; or

(ii) Definitive evidence concerning the regional and local characteristics of the fault justifies use of a different value. For example, thrust or bedding-plane faults may require an increase in width of the zone to account for the projected dip of the fault plane; or

(iii) More detailed three-dimensional information, such as that obtained from precise investigative techniques, may justify the use of a narrower zone. Possible examples of such techniques are the use of accurate records from closely spaced drill holes or from closely spaced, high-resolution offshore geophysical surveys.

In delineating the zone requiring detailed faulting investigation for a fault, the center of the zone shall coincide with the center of the fault at the point of nearest approach of the fault to the nuclear power plant as illustrated in Figure 1.

(c) *Determination of Design Bases for Seismically Induced Floods and Water Waves.* The size of seismically induced floods and water waves which could affect a site from either locally or distantly generated seismic activity shall be determined, taking into consideration the results of the investigation required by paragraph (c) of section IV. Local topographic characteristics which might tend to modify the possible runup and drawdown at the site shall be considered. Adverse tide conditions shall also be taken into account in determining the effect of the floods and waves on the site. The characteristics of the earthquake to be used in evaluating the offshore effects of local earthquakes shall be determined by a procedure similar to that used to determine the characteristics of the Safe Shutdown Earthquake in paragraph V(a).

(d) *Determination of Other Design Conditions.*—(1) *Soil Stability.* Vibratory ground motion associated with the Safe Shutdown Earthquake can cause soil instability due to ground disruption such as fissuring, differential consolidation, liquefaction, and cratering which is not directly related to surface faulting. The following geologic features which could affect the foundations of the proposed nuclear power plant structures shall be evaluated, taking into account the information concerning the physical properties of materials underlying the site developed pursuant to paragraphs (a)(1), (3), and (4) of Section IV and the effects of the Safe Shutdown Earthquake:

(i) Areas of actual or potential surface or subsurface subsidence, uplift, or collapse resulting from:

(a) Natural features such as tectonic depressions and cavernous or karst terrains, particularly those underlain by calcareous or other soluble deposits;

(b) Man's activities such as withdrawal of fluid from or addition of fluid to the subsurface, extraction of minerals, or the loading effects of dams or reservoirs; and

(c) Regional deformation

(ii) Deformational zones such as shears, joints, fractures, folds, or combinations of these features.

(iii) Zones of alteration or irregular weathering profiles and zones of structural weakness composed of crushed or disturbed materials.

(iv) Unrelieved residual stresses in bedrock.

(v) Rocks or soils that might be unstable

because of their mineralogy, lack of consolidation, water content, or potentially undesirable response to seismic or other events. Seismic response characteristics to be considered shall include liquefaction, thixotropy, differential consolidation, cratering, and fissuring.

(2) *Slope stability.* Stability of all slopes, both natural and artificial, the failure of which could adversely affect the nuclear power plant, shall be considered. An assessment shall be made of the potential effects of erosion or deposition and of combinations of erosion or deposition with seismic activity, taking into account information concerning the physical property of the materials underlying the site developed pursuant to paragraph (a)(1), (3), and (4) of Section IV and the effects of the Safe Shutdown Earthquake.

(3) *Cooling Water Supply.* Assurance of adequate cooling water supply for emergency and long-term shutdown decay heat removal shall be considered in the design of the nuclear power plant, taking into account information concerning the physical properties of the materials underlying the site developed pursuant to paragraphs (a)(1), (3), and (4) of section IV and the effects of the Safe Shutdown Earthquake and the design basis for surface faulting. Consideration of river blockage or diversion or other failures which may block the flow of cooling water, coastal uplift or subsidence, or tsunami runup and drawdown, and failure of dams and intake structures shall be included in the evaluation, where appropriate.

(4) *Distant Structures.* Those structures which are not located in the immediate vicinity of the site but which are safety related shall be designed to withstand the effect of the Safe Shutdown Earthquake and the design basis for surface faulting determined on a comparable basis to that of the nuclear power plant, taking into account the material underlying the structures and the different location with respect to that of the site.

VI APPLICATION TO ENGINEERING DESIGN

(a) *Vibratory Ground Motion.*—(1) *Safe Shutdown Earthquake.* The vibratory ground motion produced by the Safe Shutdown Earthquake shall be defined by response spectra corresponding to the maximum vibratory accelerations at the elevations of the foundations of the nuclear power plant structures determine pursuant to paragraph (a)(1) of Section V. The response spectra shall relate the response of the foundations of the nuclear power plant structures to the vibratory ground motion, considering such foundations to be single-degree-of-freedom damped oscillators and neglecting soil-structure interaction effects in view of the limited data available on vibratory ground motions of strong earthquakes. It usually will be appropriate that the response spectra be smoothed design spectra developed from a series of response spectra related to the vibratory motions caused by more than one earthquake.

The nuclear power plant shall be designed so that, if the Safe Shutdown Earthquake occurs, certain structures, systems, and components will remain functional. These structures, systems, and components are those necessary to assure (i) the integrity of the reactor coolant pressure boundary, (ii) the capability to shut down the reactor and maintain it in a safe condition, or (iii) the capability to prevent or mitigate the consequences of accidents which could result in potential offsite exposures comparable to the guideline exposures of this part. In addition to seismic loads, including aftershocks, applicable concurrent functional and accident-induced loads shall be taken into account in the design of these safety-related structures, systems, and components. The design of the nuclear power plant shall also take into account the possible effects of the Safe

Shutdown Earthquake on the facility foundations by ground disruption, such as fissuring, differential consolidation, cratering, liquefaction, and landsliding, as required in paragraph (d) of section V.

The engineering method used to ensure that the required safety functions are maintained during and after the vibratory ground motion associated with the Safe Shutdown Earthquake shall involve the use of either a suitable dynamic analysis or a suitable qualification test to demonstrate that structures, systems and components can withstand the seismic and other concurrent loads, except where it can be demonstrated that the use of an equivalent static load method provides adequate conservatism.

The analysis or test shall take into account soil-structure interaction effects and the expected duration of vibratory motion. It is permissible to design for strain limits in excess of yield strain in some of these safety-related structures, systems, and components during the Safe Shutdown Earthquake and under the postulated concurrent conditions, provided that the necessary safety functions are maintained.

(2) *Operating Basis Earthquake.* The Operating Basis Earthquake shall be defined by response spectra. All structures, systems, and components of the nuclear power plant necessary for continued operation without undue risk to the health and safety of the public shall be designed to remain functional and within applicable stress and deformation limits when subjected to the effects of the vibratory motion of the Operating Basis Earthquake in combination with normal operating loads. The engineering method used to ensure that these structures, systems, and components are capable of withstanding the effects of the Operating Basis Earthquake shall involve the use of either a suitable dynamic analysis or a suitable qualification test to demonstrate that the structures, systems and components can withstand the seismic and other concurrent loads, except where it can be demonstrated that the use of an equivalent static load method provides adequate conservatism. The analysis or test shall take into account soil-structure interaction effects and the expected duration of vibratory motion.

(3) *Required Seismic Instrumentation.* Suitable instrumentation shall be provided so that the seismic response of nuclear power plant features important to safety can be determined promptly to permit comparison of such response with that used as the design basis. Such a comparison is needed to decide whether the plant can continue to be operated safely and to permit such timely action as may be appropriate.

These criteria do not address the need for instrumentation that would automatically shut down a nuclear power plant when an earthquake occurs which exceeds a predetermined intensity. The need for such instrumentation is under consideration.

(b) *Surface Faulting.* (1) If the nuclear power plant is to be located within the zone requiring detailed faulting investigation, a detailed investigation of the regional and local geologic and seismic characteristics of the site shall be carried out to determine the need to take into account surface faulting in the design of the nuclear power plant. Where it is determined that surface faulting need not be taken into account, sufficient data to clearly justify the determination shall be presented in the license application.

(2) Where it is determined that surface faulting must be taken into account, the applicant shall, in establishing the design basis for surface faulting on a site take into account evidence concerning the regional and local geologic and seismic characteristics of the site and from any other relevant data.

(3) The design basis for surface faulting shall be taken into account in the design of the nuclear power plant by providing rea-

sonable assurance that in the event of such displacement during faulting certain structures, systems, and components will remain functional. These structures, systems, and components are those necessary to assure (i) the integrity of the reactor coolant pressure boundary, (ii) the capability to shut down the reactor and maintain it in a safe shutdown condition, or (iii) the capability to prevent or mitigate the consequences of accidents which could result in potential off-site exposures comparable to the guideline exposures of this part. In addition to seismic loads, including aftershocks, applicable concurrent functional and accident-induced loads shall be taken into account in the design of such safety features. The design provisions shall be based on an assumption that the design basis for surface faulting can occur in any direction and azimuth and under any part of the nuclear power plant, unless evidence indicates this assumption is not appropriate, and shall take into account the estimated rate at which the surface faulting may occur.

(c) *Seismically Induced Floods and Water Waves and Other Design Conditions.* The design basis for seismically induced floods and water waves from either locally or distantly generated seismic activity and other design conditions determined pursuant to paragraphs (c) and (d) of Section V, shall be taken into account in the design of the nuclear power plant so as to prevent undue risk to the health and safety of the public.

FIGURE 1—DIAGRAMMATIC ILLUSTRATION OF DELINEATION OF WIDTH OF ZONE REQUIRING DETAILED FAULTING INVESTIGATIONS FOR SPECIFIC NUCLEAR POWER PLANT LOCATION.

APPENDIX B

CALCULATION OF DISTANCE FACTORS FOR POWER
AND TEST REACTOR SITES*

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I. PURPOSE

It is the intent that this document to provide reference information and guidance on procedures and basic assumptions whereby certain factors pertinent to reactor siting as set forth in Title 10 Code of Federal Regulations Part 100 (10 CFR 100)⁽¹⁾ can be used to calculate distance requirements for reactor sites which are generally consistent with current siting practices.

For any proposed reactor: the performance experience accumulated elsewhere; the engineering safeguards; the inherent stability and safety features; and the quality of design, materials, construction, management and operation are all important factors that must be included in the evaluation of the suitability of a site.

For a particular site; size, topography, meteorology, hydrology, ease of warning and removing people in times of emergency, and thoroughness of plans and arrangements for minimizing injuries and interference with offsite activities, all enter an evaluation.

Consideration of these as well as other aspects of hazards evaluation involves so many different situations and such complex technological problems

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*Second printing--minor editorial changes made from the first printing.

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that it would be quite impossible to anticipate and answer all questions that will arise.

This technical document sets forth one method of computing distances and exposures, for one general class of reactors. In developing this example conservative assumptions have been intentionally selected.

Designers of reactors are expected to examine all significant aspects of the hazards and safety problem they believe are appropriate to the particular situation with which they are dealing. In any case, the designer and/or applicant bears the responsibility for justifying all the assumptions and methods of calculation used in a hazards evaluation. The fact that aspects of the problem are not considered in the example set forth here, does not in any way relieve the designer and/or applicant of the responsibility for carefully examining, in his particular case, every significant facet of the hazards and safety problem.

II. INTRODUCTION

An applicant for a license to construct a power or test reactor is required by Atomic Energy Commission (AEC) regulations, Title 10 Code of Federal Regulations Part 50 (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. Approval or disapproval is given by the Commission after review and evaluation of the reactor design and the proposed location by the Division of Licensing and Regulation and the Advisory Committee on Reactor Safeguards (ACRS). Such review and evaluation includes an analysis of the consequences of potential accidents.

The probability and consequences of major 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, "Theoretical Possibilities and Consequences of Major Accidents in Large Nuclear Power Plants",⁽²⁾ that the probability of a major accident in reactor plants constructed and operated in accordance with general practices now observed 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."

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. The question of suitability of a site for a reactor, however, requires consideration not only of the factors influencing the probability of occurrence of an accident, but also the risk in terms of possible exposure of people to the hazardous consequences of such an accident. Although the probability of a serious accident may be primarily a function of facility design and the risk in terms of exposure may be primarily a function of location, the two are not independent. Site characteristics may dictate the inclusion of specific engineered safeguard features and a proposed facility design in turn may have marked influence on the acceptability of the site for location of the reactor.

Values of radiation exposure dose that can be used as reference values in the evaluation of reactor sites have been set forth in 10 CFR 100. Considerations that led to the establishment of these reference values and the site criteria in which they are embodied are discussed in the sections that follow. In addition, a hypothetical case is analyzed to illustrate the calculation of distance factors as required by 10 CFR 100.

III. BASIC CRITERIA

The AEC has set forth in 10 CFR 100 a number of the factors considered by the Commission in the evaluation of reactor sites and the general criteria used at this time as guides in approving or disapproving proposed sites. One of the factors identified is the following:

"Population density and use characteristics of the site environs, including, among other things, the exclusion area, low population zone, and population center distance."

The guides (10 CFR 100.11) also set forth pertinent factors to be considered in estimating the exclusion area, low population zone and population center distance.

Specifically, 10 CFR 100 requires an applicant for a construction permit to determine the following:⁽¹⁾

- "(1) An exclusion area of such size that an individual located at any point on its boundary for two hours immediately following onset of the postulated fission product release would not receive a total radiation dose to the whole body in excess of 25 rem or a total radiation dose in excess of 300 rem to the thyroid from iodine exposure.
- (2) A low population zone of such size that an individual located at any point on its outer boundary who is exposed to the radioactive cloud resulting from the postulated fission product release (during the entire period of its passage) would not receive a total radiation dose to the

whole body in excess of 25 rem or a total radiation dose in excess of 300 rem to the thyroid from iodine exposure.

- (3) A population center distance of at least $1 \frac{1}{3}$ times the distance from the reactor to the outer boundary of the low population zone. In applying this guide due consideration should be given to the population distribution within the population center. Where very large cities are involved, a greater distance may be necessary because of total integrated population dose considerations."

In these criteria, two concepts are worthy of particular comment:

Note 1: Exposure Limits

The whole body dose of 25 rem referred to in the above excerpts from 10 CFR 100 corresponds numerically to the once in a lifetime accidental or emergency dose for radiation workers, which, according to NCRP recommendations⁽³⁾, may be disregarded in the determination of their radiation exposure status. However, neither its use in the context of this regulation nor that of a correspondingly low internal organ dose (such as, for example, the 300 rem to the thyroid might be considered in this application) is intended to imply that these numbers constitute acceptable emergency doses to the public under accident conditions. Rather, this 25 rem value and the 300 rem thyroid value have been set forth in these guides as reference values which can be used in the evaluation of reactor sites for reactors that reflect through their design, construction and operation an exceedingly low probability for a major accident, and through location and other safeguards against the hazardous consequences of an accident, should one occur, a low probability of public damage from such accidents. These exposure values cannot be considered as being independent from the likelihood of serious accidents nor from considerations of the total number of persons that might be exposed.

They have been set forth as reasonable bases for reactor site evaluations in the context of considerations such as those indicated in Section V. of this document.

Note 2: Population Center Distance

One basic objective of the criteria is to assure that the cumulative exposure dose to large numbers of people as a consequence of any nuclear accident should be low in comparison with what might be considered reasonable for total population dose. Further, since accidents of greater potential hazard than those commonly postulated as representing an upper limit are conceivable, although highly improbable, it was considered desirable to provide for protection against excessive exposure doses of people in large centers, where effective protective measures might not be feasible. Neither of these objectives were readily achievable by a single criterion. Hence, the population center distance was added as a site requirement when it was found for several projects evaluated that the specifications thereof would approximately fulfill the desired objectives and reflect a more accurate guide to current siting practices. In an effort to develop more specific guidance on the total man-dose concept, the Commission intends to give further study to the subject. Meanwhile, in recognition of the problem, the population center distance to very large cities may have to be greater than those suggested by these guides.

IV. POSTULATION OF A MAJOR REACTOR ACCIDENT

In evaluating proposed reactor sites, the basic safety questions involve the possibility of accidents which might cause radioactivity release to areas beyond the site, the possible magnitudes of such releases and the consequences these might have. Practically, there are two difficult aspects to the estimation of potential accidents in a proposed reactor which affect the problem of site evaluation.

1. The necessity for site appraisal arises early in the life of a project when many of the detailed features of design which might affect the accident potential of a reactor are not settled.
2. The inherent difficulty of postulating an accident representing a reasonable upper limit of potential hazard.

In practice, after systematic identification and evaluation of foreseeable types of accidents in a given facility, a nuclear accident is then postulated which would result in a potential hazard that would not be exceeded by any other accident considered credible during the lifetime of the facility. Such an accident has come to be known as the "maximum credible accident".

For pressurized and boiling water reactors, for example, the "maximum credible accident" has frequently 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. There may be other combinations of events which could also release significant

amounts of fission products to the environment, but in every case, for the events described above to remain the maximum credible accident the probability of their occurrence should be exceedingly small, and their consequences should be less than those of the maximum credible accident. In the analysis of any particular site-reactor combination, a realistic appraisal of the consequences of all significant and credible fission release possibilities is usually made to provide an estimate in each case of what actually constitutes the "maximum credible" accident. This estimated or postulated accident can then be evaluated to determine whether or not the criteria set out in 10 CFR 100 are met. As a further important benefit, such systematic analyses of potential accidents often lead to discovery of ways in which safeguards against particular accidents can be provided.

Since a number of analyses have indicated that the pipe rupture-meltdown sequence in certain types of water cooled reactors would result in the release of fission products not likely to be exceeded by any other "credible" accident, this accident was designated the "maximum credible accident" (MCA) for these reactors. The remainder of this discussion will refer chiefly to this type of reactor and this type of accident. Corresponding maximum credible accidents can by similar analyses be postulated for gas-cooled, liquid metal cooled, and other types of reactors.

Power and test reactors presently being operated or constructed near inhabited areas, pursuant to licenses issued by the Commission, are enclosed within external containment vessels of some type. 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 breaching of the two inner barriers, the fuel cladding and the primary system, and then the external barrier, the containment building.

A gross release of fission products into the reactor building might be initiated by: (1) a nuclear power excursion which would cause pressures in the primary cooling system sufficient to rupture the coolant piping or some part of the system; or (2) a failure of the piping or other parts of the system due to some defect in the materials. In either case, loss of the coolant would set the stage for possible fuel meltdown from the heat of fission product decay.

The rupture of the coolant system from high internal pressures due to uncontrolled internal heat generation would only occur after such failures as the following:

- 1) reactivity control mechanisms fail to function;
- 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 spontaneous 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 occurs and coolant is lost, fuel melting would require that:

- 1) decay heat is sufficient to increase fuel element temperature to the melting point, and
- 2) safeguard systems usually provided to flood or spray the core with water are either inoperative or insufficient to keep fuel elements from melting.

From such considerations, and from detailed analysis of the inherent self-stabilizing characteristics and engineered "accident prevention" safeguards, assurance is obtained that the likelihood of a major reactor accident is extremely small. Yet such a possibility for a serious accident cannot be completely discounted and the consequences, therefore, must be considered.

If a major release of fission products to the environment should occur, the potential exposure doses to persons off-site are extremely difficult to determine with exactness because of the complex and interwoven technical parameters involved. 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, but the quantity of this material that would be released as a result of an accident is unpredictable. Quantities in the order of 10 per cent of the gross activity have been assumed in the past. Experimental data would indicate these values to be conservative for accidents of the type usually visualized. The exact release can vary so much with the reactor system and with the detailed nature of an accident that the degree of conservatism in the assumptions made in any given case, is not known. Further, there is a multiplicity of possible combinations of the physical and chemical form of the radioactive materials released into the containment vessel and of the ways that atmospheric conditions might cause these radioactive materials to be transported to regions beyond the site boundary.

In accidents of the "maximum credible" type, it is usually assumed that the radioactive materials, along with erosion and corrosion products, would be dispersed in the coolant through melting or rupture of fuel elements, and then find passage to the outer containment barrier through breaches in the coolant system. On breaching, the expansion to a larger volume and a lower pressure in the containment vessel would result in steam, in addition to the gaseous fission products, and production of vapors as well as liquid and solid aerosols of a wide range of sizes. Some ejected materials may conceivably burn on contact with air, and thus increase the volatiles and fractions of fine particles. At the same time, a certain amount of the airborne fission products would be removed by such phenomena as adsorption, deposition, plate-out and steam condensation within the reactor building

or containment structure. The removal process would be complicated by conversion of normally gaseous fission products into solids by decay, and condensation of volatiles. Removal by adsorption and settling processes would be affected by turbulence. Superimposed upon these factors is the radioactive decay resulting in reduction of source strength with time by conversion to more stable isotopes. All of these factors make it difficult to determine with any exactness the radioactive content of the air which might leak out of the containment vessel.

The objective of estimating the radioactive inventory within the outer containment barrier is to attain a starting point for calculating the potential radiological hazard in the surrounding environs. For people in the proximity of the reactor building, the confined radioactive inventory represents a decaying source of direct gamma radiation which is attenuated by such factors as the structural shielding, distance, and shielding by the topography. For those at more distant points, the transport by air of the radioactive materials which might leak from the containment vessel is the major radiological consideration. For air transport, factors such as the physical 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 in determining the extent of these potential hazards. The meteorological factors will be a function of the region in which the reactor is located as well as the time of the day and season. Finally, when estimates have been made of the potential concentration of radioactivity likely to result at any distant point from the "maximum credible accident", there still remains a difficult problem of translating atmospheric concentrations into whole body or thyroid exposure doses to people at these points. For internal doses, the controlling ones, there are assumptions to be made about rates of breathing, percentage retention in the body, and cumulative doses to internal organs

resulting from retained materials. As the last exercise, there is the problem of establishing some acceptable exposure dose criteria, within the context of this procedural operation, for a comparative measure of the acceptability or unacceptability of the estimated exposures resulting from the hypothetical accident. It is from a study of these complex interwoven technical parameters that the values for the exclusion area, low population zone and population center distance must be determined.

V. ANALYTICAL METHOD

In the procedural method described herein for calculating reactor distances for power and test reactors, the highly complex phenomena involving parameters which may vary over wide ranges of values have been made manageable by simplifying assumptions, specifying that certain secondary factors are to be ignored, and fixing the values of certain key parameters. In utilizing this method, it is recognized that:

- 1) there is a substantial degree of judgment involved in establishing the basic assumptions and assigning definitive values to variable parameters;
- 2) the results obtained are 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 and fixed with certainty--an impossibility in the present state of the art;
- 3) the net effect of the assumptions and approximations is believed to give more conservative results (greater distances) than would be the case if more accurate calculations could be made.

While this approach represents a considerable simplification in the handling of the many complex phenomena involved, it represents the same very conservative approach to site selection that has characterized such evaluations in the past.

A. Fundamental Assumptions

The fundamental assumptions upon which the distances are calculated with estimates of the degree of conservatism represented in each case are as follows.

1. 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.⁽²⁾ The probability is still lower for an accident in which significant amounts of fission products are released into the confined primary coolant system and a great deal lower for accidents which would release significant quantities of radioactivity from the primary system into the reactor building.
2. It is assumed that the reactor is a pressurized water type for which the maximum credible accident will release into the reactor building 100 percent of the noble gases, 50 percent of the halogens and 1 percent of the solids in the fission product inventory. Such a release represents approximately 15 percent of the gross fission product activity.⁽¹¹⁾
3. Fifty percent of the iodines in the containment vessel is assumed to remain available for release to the atmosphere. The remaining fifty percent of the iodines is assumed to

absorb onto internal surfaces of the reactor building or adhere to internal components. Rather than the assumed reduction factor of two, it is estimated that removal of airborne iodines by various physical phenomena such as adsorption, adherence and settling could give an effect of 3-10 reduction in the final result. Credit has not been taken for the effects of washdown or filtering from protective safeguards such as cooling sprays and internal air recirculating systems. Washdown features and filtering networks could provide additional reduction factors of 10-1000.

4. The release of available (airborne) radioactivity from the reactor building to the environment is assumed to occur at a constant leakage rate of 0.1 per cent per day. The leakage and pressure conditions are assumed to persist throughout the effective course of the accident, which for practical purposes, would be until the iodine activity becomes insignificant. The maximum pressure within the reactor building and the leakage rate would actually 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, it is estimated that the final off-site doses calculated may be too high by factors of 5-10.
5. Atmospheric dispersion of material from the reactor building is assumed to occur according to the well-known relationship developed by O. G. Sutton⁽⁴⁾ involving meteorological factors of wind velocity, atmospheric stability, and diffusion parameters. Application of this treatment to reactor hazards analysis was discussed in WASH-740,⁽²⁾ and AECU-3056.⁽⁵⁾ Recently a simplified method of dispersion calculation has been proposed, by Pasquill⁽⁶⁾ and Meade,⁽⁷⁾ which reflects recent dispersion field trials, as well as current dispersion

theories. In the hypothetical situation examined here the latter method gives the same numerical results as the Sutton method for distances out to about seven miles. Beyond this distance, the new method predicts somewhat greater concentrations.

6. The assumption is made that a shift in wind direction does not occur for the duration of the leakage of the fission products from the containment barrier. If leakage from the containment barrier is assumed to occur over a significant time period, (in the order of days) a reduction factor of 2-50 could result from shifts in wind directions. Wind meandering from any one center-line direction might also result in a reduction factor of approximately 3.
7. Atmospheric dispersion is assumed to occur under inversion type weather conditions. For weather conditions which exist for 75 percent or so of the time at most sites, the atmospheric dispersion conditions could be more favorable, by factors of 5-1000.⁽⁸⁾
8. Cloud depletion as ground deposition (particulate fallout) is not assumed during cloud travel. Such deposition during cloud travel could reduce the low population zone distance by factors of 2-5.
9. In calculating the direct gamma dose, credit is not taken for shielding by the containment structure and applicable reactor shielding or topography. In some cases it is recognized that such shielding could reduce the direct gamma dose by a factor of 2-1000.
10. Decay of fission products is assumed while they are confined to the containment building but is not assumed during their transit to the receptor point. The decay enroute is not significant for the conditions of release considered here but would lower the calculated doses slightly if included.

11. In determining the whole body direct gamma dose, only the external gamma dose due to the fission products contained in the reactor building was considered significant for the assumed conditions. The whole body direct gamma dose due to the cloud passage for the assumed conditions would contribute only on the order of 1 to 10 percent of the total whole body direct gamma dose at the exclusion and low population zone distances.⁽⁵⁾

Thus, even if the postulated maximum credible accident should occur, the resulting exposure doses would probably be many times lower than those calculated by the indicated method.

On the other hand, there are potential, conceivable conditions which would result in larger fission product releases than those assumed to be released in the maximum credible accident, and the consequences could be more hazardous. Other potentially more hazardous factors than those represented by the example calculation include the following conditions.

1. Total radioactivity release to the containment vessel could theoretically be up to six times as large as those assumed. Release of long-lived fission products to the containment vessel could theoretically be up to 99 times as large as that assumed. Such releases would increase doses to the lung, bone, and total body.
2. For some sites, the atmospheric diffusion conditions for a small proportion of time could be worse than those assumed in these calculations. Such diffusion conditions could result in an increase in the inhalation doses.
3. If the external containment structure should be rendered completely ineffective at the outset of the accident, the

consequences of the "maximum credible" accident would be increased many orders of magnitude. In such a case, the dose from the cloud and ground contamination could become significant in determining the external dose.

Although the analytical approach presented herein does not take into account the effects of the full play of all the variables and influencing factors, it is considered to be a reasonable procedure that results in distances roughly reflecting current siting practices for water-cooled reactors. The assumptions made can be used as a point of departure for consideration of particular site requirements resulting from evaluation of the characteristics of a particular reactor, its purpose, and the proposed plan of operation.

B. Inhalation Dose Calculations

The potential doses to the critical organs as a consequence of inhalation of a portion of the passing cloud were determined in the manner indicated below. For the specific conditions of this example, the thyroid dose is controlling and although the method is quite general, the results of the calculation are specific for the iodine release. If the type and conditions of release were different, the controlling dose could be that to the lung, bone, gut, or other critical organ.

The amount of radioactive material inhaled by a person standing a distance, d (meters), downwind for time, τ (seconds), on the centerline of a cloud of radioactive material being continuously emitted from a ground level source is given by equation (1).

$$A_T = \frac{R Q P_0}{\pi d \alpha_y \sigma_z} \text{ curies} \dots \dots \dots (1)$$

where:

A_T is the amount of radioactive material inhaled from the cloud, (curies), during exposure for T seconds.

R is the breathing rate, ($\text{meter}^3 \cdot \text{second}^{-1}$).

Q_T is the amount of radioactive material in the total cloud, per megawatt reactor power, as it passes the receiver point d meters downwind, ($\text{curies} \cdot \text{Mw}^{-1}$), during the time interval T .

P_0 is the rated reactor power level, (Megawatts).

$\bar{u}$ is the average wind speed, ($\text{meters} \cdot \text{second}^{-1}$).

σ_y, σ_z are standard deviations of the cloud centerline concentrations in the vertical and horizontal directions, respectively.*

$$\sigma_y = \frac{1}{\sqrt{2}} C_y d^{1-n/2}, \quad \sigma_z = \frac{1}{\sqrt{2}} C_z d^{1-n/2}$$

C_y, C_z are the virtual diffusion coefficients in the vertical and horizontal planes, respectively, ($\text{meters}^{n/2}$).

n is the stability parameter, (dimensionless).

d is the distance downwind, (meters).

Equation (1) is the time integrated expression resulting from the O. G. Sutton model of atmospheric diffusion, neglecting depletion of the cloud either by radioactive decay or scavenging during transit, multiplied by the breathing rate.

Meteorological parameters were selected to be indicative of slow dispersion at a rate estimated to occur at a reasonable frequency. Such conditions could be expected to apply between 15 percent and 25 percent of the time in most areas of the United States. They would correspond closely to Pasquill's type F, stable dispersion regime, which has a frequency of

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*See Appendix A for further discussion.

occurrence (in England) in this range, according to Beattie.⁽⁹⁾

parameter values used were:

$$\bar{u} = 1 \text{ meter} \cdot \text{sec}^{-1}$$

$$C_y = 0.40 \text{ meters}^{n/2}$$

$$C_z = 0.07 \text{ meters}^{n/2}$$

$$n = 0.5$$

$$\sigma_y = \left[\frac{1}{2} C_y^2 d^{2-n} \right]^{1/2} = \frac{0.40}{\sqrt{2}} d^{0.75}$$

$$\sigma_z = \left[\frac{1}{2} C_z^2 d^{2-n} \right]^{1/2} = \frac{0.07}{\sqrt{2}} d^{0.75}$$

The "source term", Q_i , in equation (1) will be dependent upon the amount of radioactive material which has accumulated in the reactor during operation. A simplified formula for the reactor inventory, q_i , for a specific isotope is given by equation (2).

$$q_i = \frac{P_0 \times 3.2 \times 10^{16} \times Y_i (1 - e^{-\lambda_i T_0})}{3.7 \times 10^{10}}$$

$$q_i = 0.865 \times 10^6 P_0 Y_i (1 - e^{-\lambda_i T_0}) \text{ (curies)} \dots \dots \dots (2)$$

where:

q_i is the amount of isotope type i contained by the reactor at shutdown, (curies).

P_0 is the rated reactor power level, (Megawatts).

3.2×10^{16} is the number of fissions $\cdot \text{second}^{-1} \cdot \text{megawatt}^{-1}$.

Y_i is the fission yield, (atoms $_i \cdot \text{fission}^{-1}$).

λ_i is the radiological decay constant for the isotope, equal to $\frac{0.693}{T_i}$, (seconds $^{-1}$).

T_i is the radiological half-life for the isotope, (seconds $^{-1}$).

T_0 is the time interval during which the reactor has operated, (seconds).

3.7×10^{10} is the number of disintegrations $\cdot \text{sec}^{-1} \cdot \text{curie}^{-1}$.

When the reactor has been operated for a time interval such that $T_0 \gg T_r$, the term $e^{-\lambda_r T_0}$ becomes insignificant and the resulting formula for the "saturation" inventory, q_s , is given by equation (3).

$$q_s = 0.865 \times 10^6 P_0 Y_i \text{ (curies)} \dots \dots \dots (3)$$

Note that this is only true when $T_0 \gg T_r$, and therefore does not hold for very long-lived isotopes. The approximation is adequate for iodines but inadequate for Sr-90. Saturation values for the several iodine isotopes per Megawatt are given in Table I.

Table I. Saturation Inventory of Iodine Isotopes

<u>Isotope</u>	$\lambda_r^{(10)}$ (sec^{-1})	<u>Yield⁽¹¹⁾</u> (%)	<u>$[q_s/P]$</u> (curies/Mw)
131	9.96×10^{-7}	2.9	2.51×10^4
132	8.26×10^{-5}	4.4	3.81×10^4
133	9.20×10^{-6}	6.5	5.63×10^4
134	2.20×10^{-4}	7.6	6.58×10^4
135	2.86×10^{-5}	5.9	5.10×10^4

The amount of a specific isotope, Q_r , per Megawatt power, which is released from the reactor building to the atmosphere during the time interval, T , assuming constant leak rate and radioactive decay only until release, is given by equation (4).

$$Q_r = P_p P_b \left[\frac{q_s}{P} \right] \int_0^T \lambda_1 e^{-(\lambda_1 + \lambda_r)t} dt \text{ (curies-Mw}^{-1}\text{)}$$

$$Q_r = P_p P_b \left[\frac{q_s}{P} \right] \frac{\lambda_1}{\lambda_1 + \lambda_r} [1 - e^{-(\lambda_1 + \lambda_r)T}] \text{ (curies-Mw}^{-1}\text{)} \dots \dots (4)$$

Where:

P is unit reactor power (one megawatt).

F_p is the fraction of the isotope released from the primary containment system to the building.

F_b is the fraction of the isotope which remains airborne and available to be released from the building to the atmosphere.

q_s is given by equation (2).

λ_1 is the rate of leakage from the reactor building to the atmosphere, (seconds⁻¹).

λ_r is the radiological decay constant, (seconds⁻¹).

t is the time interval since the start of release during which exposure is assumed to take place, (seconds).

Consideration is given to a reactor which has been operated for a sufficiently long time period that saturation values, q_s , for the iodine isotopes may be assumed in equation (4). Furthermore, because the radii for establishing the limit of the exclusion area and the low population zone are determined by the doses resulting from two hour and infinite exposure, respectively, t may be assumed to be 7200 seconds and infinity.

Two forms of the equation are therefore necessary for the evaluation.

For exclusion distance:

$$Q_T = F_p \times F_b \times \left[\frac{q_s}{P} \right] \frac{\lambda_1}{\lambda_1 + \lambda_r} \left[1 - e^{-(\lambda_1 + \lambda_r)7200} \right] (\text{curies.M}^{-1}) \quad (5)$$

For low population zone distance:

$$Q_T = F_p \times F_b \times \left[\frac{q_s}{P} \right] \frac{\lambda_1}{\lambda_1 + \lambda_r} (\text{curies.M}^{-1}) \quad (6)$$

The model assumed in developing equation (4) is somewhat oversimplified because it assumes that the fission product is formed directly by fission process rather than through decay. Actually, most of the iodine isotopes are formed from the decay of tellurium or, as is generally the case, from

the decay of antimony and tellurium. The actual formation is of the type $A \rightarrow B \rightarrow C$ rather than formation of C directly. When the half lives of the precursors are short compared to that of the iodine considered, the effect of the precursors may be ignored and the model is adequate, but when the half life of one or more of the precursors is long compared to the iodine isotope considered, that half life will be the controlling factor in the decay chain after shutdown and the source determination must consider this factor. In the case of I^{132} , the complete decay chain and half lives involved are:

If the reactor has been in operation sufficiently long to establish radiological equilibrium, the activity of the Te^{132} and the I^{132} are equal. Since the activity of I^{132} after reactor shutdown will be determined by the decay rate of the Te^{132} , equations (4), (5), and (6) may be used to determine the I^{132} source terms if the decay constant of the Te^{132} is used in place of the decay constant of I^{132} . A more exact determination of the source term for I^{131} would also consider that amount which would be produced from the $Sb^{131} \rightarrow Te^{131}$ chain subsequent to shutdown. The amount is relatively insignificant and the calculation would needlessly complicate the example.

Values for Q_e for the exclusion and low population distance can be readily determined because values for all the factors have been given or calculated. Table II contains a listing for each of the iodine isotopes and the two time periods involved using the values:

$$F_p = 0.5,$$

$$F_b = 0.5,$$

$$\lambda_1 = 0.1\% \text{ day}^{-1}.$$

$$= 0.001 \text{ day}^{-1},$$

$$\lambda_1 = 1.16 \times 10^{-8} \text{ sec}^{-1}.$$

The breathing rate, R , in equation (1) is also a variable. The "standard man"⁽¹⁰⁾ is considered to breathe 20 meters³.day⁻¹; half during the active 8 hours and the remaining half during his relatively inactive or resting

Table II Amount of Iodine Released in Time t

Iodine Isotope	Q_T (curies.Mw ⁻¹)	
	$t = 2$ hours (Exclusion area)	$t = \infty$ (Low population zone)
131	5.20×10^{-1}	7.20×10^1
132*	5.95×10^{-1}	4.42×10^1
133	1.15×10^0	1.75×10^1
134	6.80×10^{-1}	8.70×10^{-1}
135	9.65×10^{-1}	5.15×10^0

*includes Te¹³² contribution following shutdown for $t = \infty$.

These values of Q_T will be used directly in evaluating equation (1).

hours. Since concern for personnel in the exclusion zone is based on two hours of inhalation, consider the breathing rate to be characteristic of the active portion of the normal work day,

$$R = 10 \text{ meters}^3/8 \text{ hours} = 3.47 \times 10^{-4} (\text{m}^3.\text{sec}^{-1}).$$

For the low population zone, the average breathing rate is assumed,

$$R = 20 \text{ meters}^3/24 \text{ hrs} = 2.32 \times 10^{-4} (\text{m}^3.\text{sec}^{-1}).$$

Since values have been given or calculated for all factors in equation (1), the amount inhaled, A_T , can be determined for various distances, d , down-wind. When the amount inhaled of a specific isotope is determined, the dose to the critical organ which will be delivered by this amount can be calculated.

The dose rate, D' , to the critical organ such as the thyroid at any time subsequent to the inhalation is given by equation (7).

$$\begin{aligned}
 D' \text{ (rads. sec}^{-1}\text{)} &= A_T \text{ (curies)} \times 3.7 \times 10^{10} \text{ (dis. sec}^{-1}\text{. curie}^{-1}\text{)} \\
 &\times f_a \times \bar{E} \text{ (Mev. dis}^{-1}\text{)} 1.6 \times 10^{-6} \text{ (ergs. Mev}^{-1}\text{)} e^{-\lambda_e t} \\
 &\div m \text{ (grams)} 100 \text{ (ergs. gm}^{-1}\text{. rad}^{-1}\text{)} \\
 &= \frac{5.92 \times 10^2 A_T f_a \bar{E} e^{-\lambda_e t}}{m} \text{ (rads. sec}^{-1}\text{)} \dots (7)
 \end{aligned}$$

Where:

A_T is given by equation (1).

f_a is the fraction of the amount inhaled which is deposited in the critical organ.

$\bar{E}$ is the effective energy absorbed by the critical organ per disintegration, (Mev).

$$\lambda_e = \lambda_r + \lambda_b = \frac{0.693}{T_e} \text{ (sec}^{-1}\text{)}.$$

λ_b is the biological elimination rate for the isotope, (sec⁻¹).

T_e is the effective half life for the isotope in the body, (sec).

m is the mass of the critical organ, (grams).

And the dose to the critical organ, delivered in time, T is given by equation (8).

$$\begin{aligned}
 D_T &= \int_0^T D' dt \\
 &= \frac{5.92 \times 10^2 A_T f_a \bar{E}}{m \lambda_e} [1 - e^{-\lambda_e T}] \text{ (rads)} \\
 D_T &= \frac{8.54 \times 10^2}{m} A_T f_a \bar{E} T_e [1 - e^{-\frac{0.693 T}{T_e}}] \text{ (rads)} \dots (8)
 \end{aligned}$$

When the time, T , (over which the dose is determined) is much greater than the effective half life of the isotopes, the quantity, $e^{-\frac{0.693 T}{T_e}}$, becomes insignificant and the dose to the critical organ is given by equation (9).

$$D_{\text{in}} = 8.54 \times 10^2 \frac{A_T f_a E T_e}{\text{sec}} \text{ (rads)} \dots \dots \dots (9)$$

Using equation (9), $\frac{D_{\text{in}}}{A_T}$ has been evaluated for the iodine isotopes and the values are presented in Table III.

Table III Dose to Critical Organ Per Iodine Curie Inhaled

Iodine Isotope	T_e (sec)	D_{in}/A_T (rads.curie ⁻¹)
131	6.57×10^5	1.48×10^6
132	8.39×10^3	5.35×10^4
133	7.52×10^4	4.0×10^5
134	3.11×10^3	2.5×10^4
135	2.42×10^4	1.24×10^5

C. External Gamma Dose Calculations

The external gamma radiation dose at the exclusion and low population zone distances due to fission products contained in the reactor building were determined in the following manner. The source of radiation was considered to be those fission products released from the primary system to the containment building--krypton, xenon, iodines, and a mixture of the remaining "solid" mixed fission products.

From a point source of radiation-given off by a specific gamma emitting isotope, the dose rate at a distance, d (meters), away in air is given by equation (10).

$$\begin{aligned} \text{Dose rate, } D' \text{ (rads.sec}^{-1}\text{)} &= F_p \times P_0 \text{ (Mw)} \times \left[\frac{q_g}{P} \right] \text{ (curies.Mw}^{-1}\text{)} \\ &\times 3.7 \times 10^{10} \text{ (dis.sec}^{-1}\text{.curie}^{-1}\text{)} \times E_\gamma \text{ (Mev.dis}^{-1}\text{)} \\ &\times 1.6 \times 10^{-6} \text{ (ergs.Mev}^{-1}\text{)} \times \mu_a \text{ (meter}^{-1}\text{)} B e^{-\mu d} e^{-\lambda t} \\ &\div 1.293 \times 10^3 \text{ (grams.meter}^{-3}\text{)}_{\text{air}} \times 10^2 \text{ (ergs.gram}^{-1}\text{.rad}^{-1}\text{)} \\ &\times 4\pi d^2 \text{ (meter}^2\text{)} \dots \dots \dots (10) \end{aligned}$$

In equation (10), the dose buildup factor, B, is expressed by equation (10a).

$$B = 1 + k \mu d \dots \dots \dots (10a)$$

After combining terms, equation (10) can be expressed as

$$D = 6.15 \times 10^{-12} S_0 \times F_p \times P_0 \mu_a d^{-2} [1 + k \mu d] e^{-\mu d} \times e^{-\lambda_r t} \text{ (rads. sec}^{-1}\text{)} \dots \dots (11)$$

$$D = C e^{-\lambda_r t}$$

Where:

S_0 is the initial gamma source strength for the specific isotope at shutdown, (Mev.sec⁻¹.Mw⁻¹).

F_p is the fraction of the isotope released to the reactor building.

P_0 is the rated reactor power level, (Mw).

μ_a is the energy absorption coefficient, (meter⁻¹).

k is the linear absorption constant, ($\frac{\mu - \mu_a}{\mu_a}$).

μ is the linear absorption coefficient, (meter⁻¹).

d is the distance to receptor, (meters).

λ_r is the radiological decay constant, (sec⁻¹).

t is the time after shutdown, (sec).

C is a constant defined by the equation.

The term, S_0 , combines three terms in equation (11):

$$S_0 = \left[\frac{Q_s}{P} \right] \times 3.7 \times 10^{10} \times E_i \text{ (Mev. sec}^{-1} \cdot \text{Mw}^{-1}\text{)}.$$

Where:

$\left[\frac{Q_s}{P} \right]$ is the saturation inventory, (curies.Mw⁻¹).

E_i is the total gamma energy per disintegration, (Mev.dis⁻¹).

3.7×10^{10} is the number of disintegrations.sec⁻¹.curie⁻¹.

Table IV contains values of S_0 , F_p , S_R , T_R , μ , μ_a , k , and E_{avg} for the isotopes, the "solid" mixed fission products and the gross fission products assumed to be released to the reactor building. The values of μ , μ_a , and k are energy

Table IV. External Gamma Dose Data

Gamma Source	Half Life $T_{1/2}$	(11) "Average" Energy $E(\text{Kev})$	Initial (11) Source Strength $S_0(\text{Kev/sec-Mw})$	Release Fraction F_p	Released Source Strength $S_R(\text{Kev/sec-Mw})$	Linear (13) Absorption Coefficient $\mu(\text{meter}^{-1})$	Linear (13) Absorption Constant k	Energy (13) Absorption Coefficient $\mu_a(\text{meter}^{-1})$
I. Iodine								
131	8.05 days	0.4	3.63×10^{14}	0.5	1.81×10^{14}	1.23×10^{-2}	2.22	3.8×10^{-3}
132	2.4 hrs.	0.8	2.82×10^{15}	0.5	1.41×10^{15}	9.1×10^{-3}	1.45	3.7×10^{-3}
133	20.8 hrs.	0.55	1.15×10^{15}	0.5	0.57×10^{15}	1.08×10^{-2}	1.85	3.9×10^{-3}
134	52.5 mins.	1.3	3.10×10^{15}	.55	1.55×10^{15}	7.2×10^{-3}	1.10	3.4×10^{-3}
135	6.68 hrs.	1.5	2.90×10^{15}	0.5	1.45×10^{15}	6.7×10^{-3}	1.02	3.3×10^{-3}
Total Iodine	--	--	1.03×10^{16}	0.5	5.16×10^{15}	--	--	--
II. Xenon								
131m	12.0 days	0.163	1.50×10^{12}	1.0	1.50×10^{12}	1.7×10^{-2}	4.0	3.3×10^{-3}
133m	2.3 days	0.233	1.16×10^{13}	1.0	1.16×10^{13}	1.5×10^{-2}	3.2	3.6×10^{-3}
133	5.27 days	0.081	1.67×10^{14}	1.0	1.67×10^{14}	2.2×10^{-2}	7.0	2.7×10^{-3}
135m	15.6 mins.	0.520	2.94×10^{14}	1.0	2.94×10^{14}	1.1×10^{-2}	1.9	3.9×10^{-3}
135	9.13 hrs.	0.250	4.65×10^{14}	1.0	4.65×10^{14}	1.5×10^{-2}	3.0	3.6×10^{-3}
Total Xenon	--	--	9.40×10^{14}	1.0	9.40×10^{14}	--	--	--
III. Krypton								
83m	114 mins.	0.02	6.35×10^{12}	1.0	6.35×10^{12}	(dose considered negligible)		
85m	4.36 hrs.	0.20	8.65×10^{13}	1.0	8.65×10^{13}	1.6×10^{-2}	3.5	3.5×10^{-3}
87	78 mins.	2.00	4.84×10^{14}	1.0	4.84×10^{14}	5.8×10^{-3}	0.9	3.0×10^{-3}
88	2.77 hrs.	2.00	2.44×10^{15}	1.0	2.44×10^{15}	5.8×10^{-3}	0.9	3.0×10^{-3}
Total Krypton	--	--	3.02×10^{15}	1.0	3.02×10^{15}	--	--	--
IV. Mixed Fission Products ("Solids")	(varied)	0.7	3.72×10^{16}	0.01	3.72×10^{14}	1.0×10^{-2}	1.6	3.8×10^{-3}
V. Gross Fission Products								
Total of I, II, III, IV	--	--	5.15×10^{16}	--	9.49×10^{15}	--	--	--

"Solid" Fission Products have an effective half-life of 2.72 hours during the first 2 hours of decay after long-time operation and have an effective decay rate which follows $t^{-0.21}$ after the first 2 hours.

dependent. Hence, "average" energies, E_{avg} , were selected after reviewing the weighted spectrum for each isotope and mixtures.⁽¹¹⁾ The term S_R is the product of S_0 and F_p for the assumed release conditions.

Integrating the direct gamma dose rate, D' , over a specific exposure time yields the direct gamma dose.

$$D_i = \int_0^t D'_R dt (\text{rads})$$

$$= \int_0^t C e^{-\lambda_R t} dt$$

$$D_i = \frac{C}{\lambda_R} [1 - e^{-\lambda_R t}] (\text{rads}). \dots \dots \dots (12)$$

Considering the exclusion distance, the exposure time is 2 hours and equation (12) becomes:

$$D_i = \frac{C}{\lambda_R} [1 - e^{-7200 \lambda_R}] (\text{rads}). \dots \dots \dots (13)$$

and for the low population zone, the exposure time is 30 days, which is several half-lives for the isotopes of the noble gases and iodine. Equation (13) for these isotopes, may be written as:

$$D_i = \frac{C}{\lambda_R} (\text{rads}). \dots \dots \dots (14)$$

In the case of the "solid" mixed fission products the dose for the first two hours was considered to be decaying exponentially with a half-life of 2.72 hours ($\lambda_R = 7.05 \times 10^{-5} \text{ sec}^{-1}$) and subsequently as $t^{-0.21}$ based on interpretations of data from Blomeke and Todd.⁽¹¹⁾

For the first two hours, the dose was:

$$D_i = \frac{C}{\lambda_R} [1 - e^{-\lambda_R t_1}] (\text{rads}). \dots \dots \dots (15)$$

and for the 30 days, the dose was:

$$D_i = \frac{C}{\lambda_R} [1 - e^{-\lambda_R t_1}] + C e^{-\lambda_R t_1} \int_{t_1}^{t_2} t^{-0.21} dt$$

$$D_i = \frac{C}{\lambda_R} [1 - e^{-\lambda_R t_1}] + \frac{C e^{-\lambda_R t_1}}{0.79} [t_2^{0.79} - t_1^{0.79}] (\text{rads}). \dots \dots \dots (16)$$

Table V. External Gamma Dose Results

<u>Direct Gamma Dose (rads/Mv)</u>								
Gamma Source	<u>Exclusion Radius ($T = 2$ hours)</u>				<u>Low Population zone ($T = 30$ days)</u>			
	<u>100 m</u>	<u>300 m</u>	<u>600 m</u>	<u>1000 m</u>	<u>100 m</u>	<u>300 m</u>	<u>600 m</u>	<u>1000 m</u>
I. Iodine								
131	5.3×10^{-1}	1.24×10^{-2}	1.49×10^{-4}	-----	7.4×10^1	1.72×10^0	2.06×10^{-2}	9.15×10^{-5}
132	2.66×10^0	1.02×10^{-1}	3.01×10^{-3}	4.9×10^{-5}	5.95×10^0	2.26×10^{-1}	6.75×10^{-3}	1.08×10^{-4}
133	1.62×10^0	4.82×10^{-2}	9.25×10^{-4}	-----	2.44×10^1	7.25×10^{-1}	1.39×10^{-3}	1.05×10^{-4}
134	1.61×10^0	8.05×10^{-1}	3.85×10^{-3}	1.24×10^{-4}	2.03×10^1	1.02×10^{-1}	4.84×10^{-3}	1.55×10^{-4}
135	2.62×10^0	1.39×10^{-1}	7.75×10^{-3}	2.86×10^{-4}	1.40×10^1	7.40×10^0	4.11×10^{-2}	1.52×10^{-3}
Total Iodine	9.04×10^0	3.82×10^{-1}	1.57×10^{-2}	4.79×10^{-4}	1.2×10^2	3.51×10^0	7.47×10^{-2}	1.98×10^{-3}
II. Xenon								
131m	5.16×10^{-3}	5.16×10^{-5}	-----	-----	1.09×10^0	1.09×10^{-2}	3.24×10^{-4}	-----
133m	3.86×10^{-2}	5.71×10^{-4}	3.0×10^{-6}	-----	1.54×10^0	2.27×10^{-2}	1.18×10^{-4}	-----
133	6.73×10^{-1}	2.73×10^{-3}	1.83×10^{-6}	-----	6.1×10^1	2.5×10^{-2}	1.68×10^{-4}	-----
135m	1.55×10^{-1}	4.62×10^{-3}	8.25×10^{-5}	-----	1.55×10^{-1}	4.6×10^{-3}	8.25×10^{-5}	-----
135	1.41×10^0	2.02×10^{-2}	1.06×10^{-4}	-----	9.7×10^0	1.4×10^{-1}	7.3×10^{-4}	-----
Total Xenon	2.28×10^0	2.82×10^{-2}	1.93×10^{-4}	-----	7.35×10^1	2.03×10^{-1}	1.42×10^{-3}	-----
III. Krypton								
83m	(negligible contribution)				(negligible contribution)			
85m	2.40×10^{-1}	2.98×10^{-3}	1.14×10^{-5}	-----	8.8×10^{-1}	1.09×10^{-2}	4.2×10^{-5}	-----
87	5.4×10^{-1}	3.2×10^{-2}	2.27×10^{-3}	1.19×10^{-4}	8.25×10^{-1}	4.86×10^{-2}	3.44×10^{-3}	1.81×10^{-4}
88	3.52×10^0	2.1×10^{-1}	1.48×10^{-2}	7.75×10^{-4}	1.03×10^1	6.15×10^{-1}	4.32×10^{-2}	2.28×10^{-3}
Total Krypton	4.3×10^0	2.45×10^{-1}	1.71×10^{-2}	8.94×10^{-4}	1.2×10^1	6.75×10^{-1}	4.67×10^{-2}	2.46×10^{-3}
IV. Fission Products "Solids"								
	7.55×10^{-1}	2.52×10^{-2}	5.8×10^{-4}	6.0×10^{-6}	1.21×10^1	4.05×10^{-1}	9.25×10^{-3}	9.0×10^{-5}
V. Released Fission Products Total Dose								
	1.64×10^1	6.8×10^{-1}	3.29×10^{-2}	1.38×10^{-3}	2.18×10^2	4.79×10^0	1.32×10^{-1}	4.53×10^{-3}

Table VI. Inhalation Dose Results

<u>Inhalation Source</u>	<u>Inhalation Iodine Dose (rads/Mr)</u>							
	<u>Exclusion Radius ($\tau = 2$ hours)</u>				<u>Low Population Zone ($\tau = \infty$)</u>			
	<u>10^2m</u>	<u>10^3m</u>	<u>10^4m</u>	<u>10^5m</u>	<u>10^2m</u>	<u>10^3m</u>	<u>10^4m</u>	<u>10^5m</u>
Iodine								
131	6.02×10^0	1.94×10^{-1}	6.02×10^{-3}	1.94×10^{-4}	5.6×10^2	1.79×10^1	5.6×10^{-1}	1.79×10^{-2}
132	3.36×10^{-1}	1.08×10^{-1}	3.36×10^{-3}	1.08×10^{-4}	1.24×10^1	3.98×10^0	1.24×10^{-2}	3.98×10^{-3}
133	3.61×10^0	1.16×10^{-3}	3.61×10^{-4}	1.16×10^{-4}	3.66×10^{-1}	1.18×10^0	3.66×10^{-4}	1.18×10^{-6}
134	1.33×10^{-1}	4.28×10^{-2}	1.33×10^{-4}	4.28×10^{-5}	1.14×10^0	3.66×10^{-3}	1.14×10^{-3}	3.66×10^{-6}
135	9.4×10^{-1}	3.01×10^{-2}	9.4×10^{-4}	3.01×10^{-5}	3.35×10^0	1.07×10^{-1}	3.35×10^{-3}	1.07×10^{-4}
Total Iodine	1.10×10^1	3.55×10^{-1}	1.10×10^{-2}	3.55×10^{-4}	6.12×10^2	1.96×10^1	6.12×10^{-1}	1.96×10^{-2}

Values Assumed for Results:

$F_b = 0.5$
 $F_p = 0.5$
 $\bar{u} = 1 \text{ meter} \cdot \text{sec}^{-1}$
 $C_y = 0.40 \text{ meters}^{n/2}$
 $C_z = 0.07 \text{ meters}^{n/2}$
 $n = 0.5$
 $\lambda_1 = 0.1 \text{ percent} \cdot \text{day}^{-1}$

and since $\tau_2 \gg \tau_1$, equation (16) can be written:

$$D_f = \frac{C}{\lambda_r} \left[1 - e^{-\lambda_r \tau_1} \right] + \frac{C e^{-\lambda_r \tau_1}}{0.79} \tau_2^{0.79} (\text{rads}) \dots (17)$$

The total direct gamma dose is the sum of the doses from each of the source terms as determined by equations (13) and (15) for the exclusion area and equations (14) and (17) for the low population zone.

D. Results

The results of the calculations performed for the inhalation (iodine) dose and the external gamma dose for the exclusion area ($\tau = 2$ hours) and the low population zone ($\tau = \infty$ and 30 days, respectively) are presented in Tables V and VI. Based upon these results, initial estimates of distances for reactors of various power levels have been developed and are listed in Table VII.

Table VII. Calculated Radii for Water Cooled Reactors of Various Power Levels

Power Level (Mw _t)	Exclusion area distance (miles)	Low population zone distance (miles)	Population center distance (miles)
1500	0.88	13.3	17.7
1200	0.77	11.5	15.3
1000	0.67	10.3	13.7
900	0.63	9.4	12.5
800	0.58	8.6	11.5
700	0.53	8.2	10.9
600	0.48	7.2	9.6
500	0.43	6.5	8.7
400	0.37	5.4	7.2
300	0.31	4.5	6.0
200	0.29	3.4	4.5
100	0.25	2.2	2.9
50	0.21	1.4	1.9
10	0.13	0.5	0.7

The estimated radii for power reactors are graphically represented in Figures 1 and 2. For the exclusion distance, doses from both direct gamma radiation from the reactor building and from iodine in the cloud escaping from the reactor building were calculated, and the distance established on the basis of the effect requiring the greater isolation. Figure 1 shows the thyroid and whole body doses for various power levels.

Under the conditions assumed, the doses resulting from the inhalation of the isotopes of iodine are controlling for the low population zone distance and population center distance. However, it is possible that such may not always be the case and this should be checked for each case under consideration. The low population zone distance results from integrating the effects of iodine 131 through 135. The population center distance equals the low population zone distance increased by a factor of one-third. Figure 2 shows the relationship between the inhalation and direct dose for various power levels.

Figure 1. Exclusion Radius Determination.

Figure 2. Population Radius Determination.

VI. COMPARISON OF ANALYTICAL METHOD TO EXISTING REACTOR SITES

As an indication of how the use of the above analytical method results in distances reflecting current siting practices, the method was

Table VIII. Calculated Distances for Selected Reactors

Reactor	Power Level (MW _e)	Exclusion Area		Low Population Area	Population Center Distance	
		Calculated	Actual	Calculated	Calculated	Actual
		Distance (miles)	Distance (miles)	Distance (miles)	Distance (miles)	Distance (miles)
Dresden	630	0.50	0.50	7.4	9.9	14.0
Con. Ed.	585	0.48	0.30	7.0	9.4	17.0
Yankee	485	0.42	0.50	6.3	8.4	21.0
• PRDC	300	0.31	0.75	4.5	6.1	7.5
PWR	270	0.31	0.40	4.1	5.6	7.5
Consumers	240	0.30	0.50	3.9	5.2	135.0
• Hallam	240	0.30	0.25	3.9	5.2	17.0
Pathfinder	203	0.29	0.50	3.4	4.6	3.5
PC&E	202	0.29	0.25	3.4	4.6	3.0
• Phila.Elec.	115	0.26	0.57	2.4	3.2	21.0
NASA	60	0.22	0.50	1.6	2.1	3.0
CVTE	60	0.22	0.50	1.6	2.1	25.0
Elk River	58	0.22	0.23	1.5	2.0	20.0
VBWR	50	0.21	0.40	1.4	1.9	15.0
• Piqua	48	0.21	0.14	1.4	1.8	27.0

*NOTE: These reactors are not water moderated and are included in the table for illustrative purposes only. The distances for all reactors were based on the same assumption with respect to fission product release from the fuel and containment vessel and the subsequent dispersal events. There can be considerable differences between reactor types in the events that could result in a major accident and the releases that might be experienced. This must be examined on an individual basis for each reactor and the distances determined accordingly.

applied to a number of reactor projects that have been proposed or are currently authorized for construction. These results are given in Table VIII.

VII. APPENDIX

A. Relationship of the Sutton Diffusion Parameter and the Generalized Gaussian Parameter

The traditional form of the O. G. Sutton atmospheric diffusion equation describing the centerline concentration downwind of a continuous point source is generally written:

$$\frac{\bar{X}}{Q'} = \frac{1}{\pi \bar{u} C_y C_z d^{2-n}}$$

This equation was based on an extension of diffusion theory, an assumed homogeneous isotropic source, and an assumed three dimensional Gaussian distribution model.

When the receptor and cloud centerline are coincident with the ground level, the concentration is assumed to be doubled as a consequence of "ground reflection":

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

The diffusion coefficients, C_y and C_z are mathematical quantities which represent the diffusion capability of the atmosphere. However, Sutton and others found it necessary to determine values of C_y and C_z indirectly from data obtained through experimental field measurement. By expressing the diffusion coefficients in terms of standard deviations of the Gaussian distribution model which is assumed to describe the spacial relationship of cloud

concentration, the resulting equation may be written in the more useful form:

$$\frac{X}{Q'} = \frac{1}{\pi u \sigma_y \sigma_z}$$

where:

σ_y and σ_z are the standard deviations of the cloud concentration in the vertical and horizontal directions, respectively.

The factor of two which was introduced for "ground reflection" has been included in this equation. The equation in this form, with the Gaussian parameters, permit direct interpretation of experimental data obtained from field measurements.

The relationship between the generalized diffusion parameters ⁽¹⁴⁾ and the more familiar Sutton parameters ⁽⁴⁾ are expressed as:

$$\sigma_y = \frac{1}{\sqrt{2}} C_y d^{1-n/2}$$

$$\sigma_z = \frac{1}{\sqrt{2}} C_z d^{1-n/2}$$

In the generalized form, the parameters σ_y and σ_z are functions of distance and can be approximated directly from anemometer records if appropriate averaging techniques are supplied ⁽⁷⁾.

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C. List of Symbols and Definitions

<u>Symbol</u>	<u>Represents</u>	<u>Dimensions</u>
A_c	Amount of radioactive material inhaled by an individual during a specific time period.	curies
B	Build up factor.	-
C_y, C_z	Meteorological virtual diffusion coefficients in the vertical and horizontal planes, respectively.	meters ^{n/2}
d	Distance from the source of radiation or release point.	meters
D'	Dose rate delivered by an exposure to radiation.	rads·sec ⁻¹
D_T, D_∞	Dose delivered during time interval "T" or infinite time.	rads
D_i	Dose delivered by direct exposure to gamma radiation.	rads
$\bar{E}$	Effective energy absorbed by the critical organ per disintegration.	Mev·dis ⁻¹
E_{avg}	Average energy assumed for selecting values of μ .	Mev·dis ⁻¹
E_γ	Total gamma energy emitted per disintegration.	Mev·dis ⁻¹

List of Symbols and Definitions (Cont'd.)

<u>Symbol</u>	<u>Represents</u>	<u>Dimensions</u>
f_a	Fraction of inhaled material which is subsequently deposited in the critical organ.	-
f_b	Fraction of material released to the reactor building and available to be released to the atmosphere.	-
f_F	Fraction of inventory released from the primary system to the reactor building.	-
k	Linear absorption constant, $(\frac{\lambda - \lambda_a}{\lambda_a})$.	-
m	Mass of the critical organ.	grams
n	Meteorological stability parameter.	-
P	Rated reactor power level.	Mw _t
P_0	Unit reactor power (1 megawatt).	
q_s	Saturated reactor inventory for a specific isotope.	curies
q_t	Reactor inventory for a specific isotope after a finite operating time.	curies
Q_T	Amount of a specific isotope released to the atmosphere during a finite time interval per megawatt reactor power.	curies·Mw ⁻¹
R	Breathing rate.	meters ³ ·sec ⁻¹
S_0, S_T	Source terms at shutdown - total and released to reactor building.	Mev·sec ⁻¹ ·Mw ⁻¹
t, T	Time variables.	seconds
T_0	Reactor operating time.	seconds
T_b, T_e, T_r	Biological, effective, and radiological half-lives.	seconds
$\bar{u}$	Average wind speed.	meters·sec ⁻¹
τ, τ_1, τ_2	Exposure time intervals.	seconds
σ_y, σ_z	Standard deviations of cloud concentration.	-
γ_1	Fission yield.	nuclei·fission ⁻¹

List of Symbols and Definitions (Cont'd.)

<u>Symbol</u>	<u>Represents</u>	<u>Dimensions</u>
$\lambda_b, \lambda_e, \lambda_r$	Biological, effective, and radiological elimination and decay constants.	second ⁻¹
λ_1	Leak rate from the containments shell (reactor building)	second ⁻¹
μ, μ_a	Linear and energy absorption coefficients	meter ⁻¹

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3. "Permissible Dose From External Sources of Ionizing Radiation", NBS Handbook No. 59, w/addendum of April 1958, September 24, 1954.
4. Sutton, O. G., "A Theory of Eddy Diffusion in the Atmosphere", Proceedings of the Royal Society (London), 135A:143, 1932.
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APPENDIX C

GENERAL SITE SUITABILITY CRITERIA
FOR NUCLEAR POWER STATIONS

[Regulatory Guide 4.7*]

A. INTRODUCTION

The Energy Reorganization Act of 1974 places on the Nuclear Regulatory Commission (NRC) the responsibility for the licensing and regulation of private nuclear facilities from the standpoint of public health and safety. Paragraphs 100.10(b) and (c) of 10 CFR Part 100, "Reactor Site Criteria," require that the population density, use of the site environs, and the physical characteristics of the site, including seismology, meteorology, geology, and hydrology, be taken into account in determining the acceptability of a site for a nuclear power reactor. Seismic and geologic site criteria for nuclear power plants are provided in Appendix A to 10 CFR Part 100. Appendix A to 10 CFR Part 50 establishes the minimum requirements for the principal design criteria for water-cooled nuclear power plants; a number of these criteria are directly related to site characteristics as well as to events and conditions outside the nuclear power unit.

The National Environmental Policy Act of 1969 (NEPA) (83 Stat. 852), implemented by Executive Order 11514 and the Council on Environmental Quality's Guidelines of August 1, 1973 (38 FR 20550), requires that all agencies of the Federal Government prepare detailed environmental statements on proposed major Federal actions which can significantly affect the quality of the human environment. A principal objective of NEPA is to require the Federal agency to consider, in its decision-making process, the environmental impacts of each proposed major action and the available alternative actions.

Part 51, "Licensing and Regulatory Policy and Procedures for Environmental Protection," of Title 10, Code of Federal Regulations, sets forth the Nuclear Regulatory Commission's policy and procedures for the preparation and processing of environmental impact statements and related documents pursuant to Section 102(2)(C) of the NEPA.

The limitations on the Commission's authority and responsibility pursuant to the NEPA imposed by the Federal Water Pollution Control Act (86 Stat. 916) are addressed in an Interim Policy Statement published in the *Federal Register* on January 29, 1973 (38 FR 2679).

This guide discusses the major site characteristics related to public health and safety and environmental issues which the NRC staff considers in determining the suitability of sites for light-water-cooled (LWR) and high temperature gas-cooled (HTGR) nuclear power stations.^a The guidelines may be used by applicants in identifying suitable candidate sites for nuclear power stations. The decision that a station may be built on a specific candidate site is based on a detailed evaluation of the proposed site-plant combination and a cost-benefit analysis comparing it with alternative site-plant combinations as discussed in Regulatory Guide 4.2, "Preparation of Environmental Reports for Nuclear Power Stations."

Chapter 9 of Regulatory Guide 4.2 discusses the selection of a site from among alternative sites. Although it is recognized that planning methods^b will differ among applicants, Chapter 9 states that the applicant should present its site-plant selection process as the consequence of an analysis of alternatives whose environmental costs and benefits were evaluated and compared and then weighed against those of the proposed facility.

^aFor the purposes of this guide, nuclear power station refers to the nuclear reactor unit(s), nuclear steam supply, electric generating units, auxiliary systems, including the cooling system and structures such as docks that are located on a given site, and any new electrical transmission towers and lines erected in connection with the facilities.

^bSite selection methodologies that have been used by the nuclear power industry are described in "Nuclear Power Plant Siting: A Generalized Process," Atomic Industrial Forum, August 1974, National Environmental Studies Project, R-1578.

*This guide was initially published as a draft in September 1974.

This guide is intended to assist applicants in the initial stage of selecting potential sites for a nuclear power station. Each site that appears to be compatible with the general criteria discussed in this guide will have to be examined in greater detail before it can be considered to be a "candidate" site, i.e., one of the group of sites that are to be considered in selecting a "proposed" or "preferred" site.^c

This guide should be used only in the initial stage of site selection because it does not provide detailed guidance on the various relevant factors and format for ranking the relative suitability or desirability of possible sites. This guide provides a general set of safety and environmental criteria which the NRC staff has found to be valuable in assessing candidate site identification in specific licensing cases.

The information needed to evaluate potential sites at this initial stage of site selection is assumed to be limited to that information which may be obtained from published reports, public records, public and private agencies, and individuals knowledgeable about the locality of a potential site. Although in some cases the applicants may have conducted on-the-spot investigations, it is assumed here that these investigations would be limited to reconnaissance-type surveys at this stage in the site selection process.

The safety issues discussed include geologic/seismic, hydrologic, and atmospheric characteristics of proposed sites; potential effects on the station from accidents associated with nearby industrial transportation, and military facilities; and population distribution and densities in the site environs as they relate to protecting the general public from the potential radiation hazards of postulated serious accidents. The environmental issues discussed concern potential impacts from the construction and operation of nuclear power stations on ecological systems, water use, land use, the atmosphere, aesthetics, and socioeconomics.

This guide does not discuss details of the engineering designs required to ensure the compatibility of the nuclear station and the site or the detailed information required for the preparation of the safety analysis and environmental reports. In addition, nuclear power reactor site suitability as it may be affected by the Commission's materials safeguards and plant protection requirements for nuclear power plants is not addressed in this guide.

^cSee Chapter 9 of Regulatory Guide 4.2 for a discussion of site selection procedures. The "proposed" site submitted by an applicant for a construction permit is that site of a number of "candidate" sites which the applicant prefers and on which the applicant proposes to construct a nuclear power station.

Guidance concerning the siting of offshore nuclear stations, liquid metal fast breeder reactors (LMFBR), and advanced siting concepts such as underground sites and nuclear energy centers is not included in this guide.

A significant commitment of time and resources may be required to select a suitable site for a nuclear power station, including safety and environmental considerations, and to develop an acceptable design for that site. Site selection involves considerations of public health and safety, engineering and design, economics, institutional requirements, environmental impacts, and other factors. The potential impacts of the construction and operation of nuclear power stations on the physical and biological environment and on social, cultural, and economic features^a are usually similar to the potential impacts of any major industrial facility, but nuclear power stations are unique in the degree to which potential impacts of the environment on their safety must be considered. The safety requirements are primary determinants of the suitability of a site for nuclear power stations, but considerations of environmental impacts and public acceptance of nuclear power stations are also important and need to be evaluated.

In the site selection process, coordination between applicants for nuclear power stations and various Federal, State, and local agencies will be useful in identifying potential problem areas.

Appendices A and B of this guide summarize the important safety-related and environmental considerations for assessing the site suitability of nuclear power stations.

B. DISCUSSION

1. Geology/Seismology

Nuclear power stations must be designed to prevent the loss of safety related functions. Generally, the most restrictive safety-related site characteristics considered in determining the suitability of a site are surface faulting, potential ground motion and foundation conditions^b

^aBiological and physical environment includes geology, geomorphology, surface and groundwater hydrology, climatology, air quality, limnology, water quality, fisheries, wildlife, and vegetation. Social and cultural features include scenic resources, recreation resources, archeological/historical resources, and community resources including land use patterns. From "Development and the Environment: Legal Reforms to Facilitate Industrial Site Selection," final report by the Committee on Environmental Law, American Bar Association, February 1974.

^b"Classification, Engineering Properties and Field Exploration of Soils, Intact Rock and In Situ Masses," WASH-1301, March 1974, outlines some of the procedures used to evaluate site foundation properties.

(including liquefaction, subsidence, and landslide potential), and seismically induced floods. Criteria that describe the nature of the investigations required to obtain the geologic and seismic data necessary to determine site suitability are provided by Appendix A, "Seismic and Geologic Criteria for Nuclear Power Plants," to 10 CFR Part 100. Safety-related site characteristics are identified in Section 2.5 of Regulatory Guide 1.70, "Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants," and Regulatory Guide 1.59, "Design Basis Floods for Nuclear Power Plants." In addition to geologic and seismic evaluation for assessing seismically induced flooding potential, Section 2.4 of Regulatory Guide 1.70 and Regulatory Guide 1.59 describe hydrologic criteria, including coincident flood events that should be considered.

2. Atmospheric Extremes and Dispersion

The potential effect of natural atmospheric extremes (e.g., tornadoes^c and exceptional icing conditions^d) on the safety-related structures of a nuclear station must be considered. However, the atmospheric extremes that may occur at a site are not normally critical in determining the suitability of a site because safety-related structures, systems, and components can be designed to withstand most atmospheric extremes.

The atmospheric characteristics at a site are an important consideration in evaluating the dispersion of radioactive effluents both from postulated accidents and from routine releases in gaseous effluents.^e In addition to meeting the NRC requirements for the dispersion of airborne radioactive material, the station must meet State and Federal requirements of the Clean Air Amendments of 1970 (PL 91-604). This is unlikely to be an important consideration for nuclear power station siting unless (1) a site is in an area where existing air quality is near or exceeds the limits set under the Clean Air Amendments, (2) there is a potential for interaction of the cooling system plume with a plume containing noxious or toxic substances from a nearby facility, or (3) the auxiliary generators are operating.

^cRefer to Regulatory Guide 1.76, "Design Basis Tornado for Nuclear Power Plants." •

^dRefer to Section 2.4.7 of Regulatory Guide 1.70.

^eRoutine releases of airborne radioactive material must be kept "as low as practicable" [See 10 CFR Part 20, § 20.1(c).] The Commission has published a proposed rule for public comment (40 FR 33029) that substitutes "as low as is reasonably achievable" for the older, less precise term "as low as practicable" where it appears in NRC regulations and regulatory guides.

Section 50.34s of 10 CFR Part 50 sets forth the requirements for design objectives for equipment to control releases of radioactive material in effluents from nuclear power reactors.

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The atmospheric data necessary for adequate assessment of the potential dispersion of radioactive material from design basis accidents are described in Regulatory Guide 1.23, "Onsite Meteorological Programs." Models and assumptions used for evaluating the potential radiological consequences of certain postulated accidents are provided in Regulatory Guides 1.3, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss-of-Coolant Accident for Boiling Water Reactors;" 1.4, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss-of-Coolant Accident for Pressurized Water Reactors;" 1.5, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Steam Line Break Accident for Boiling Water Reactors;" 1.24, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Pressurized Water Reactor Radioactive Gas Storage Tank Failure;" and 1.25, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Fuel Handling Accident in the Fuel Handling and Storage Facility for Boiling and Pressurized Water Reactors." However, the atmospheric assumptions in the guides may not be appropriate for sites with unusual atmospheric conditions.

In the evaluation of potential sites, onsite atmospheric reconnaissance can determine if the atmospheric conditions at a site are adequately represented by the available atmospheric data for the area. Canyons or deep valleys frequently have atmospheric variables that are substantially different from those variables measured for the general region. Other topographical features such as hills, mountain ranges, and lake or ocean shorelines can affect the local atmospheric conditions at a site and may cause the dispersion characteristics at the site to be less favorable than those in the general area or

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Section 50.36a further provides that, in order to keep power reactor effluent releases as low as practicable, each license authorizing operation of such a facility will include technical specifications regarding the establishment of effluent control equipment and reporting of actual releases.

Appendix I to 10 CFR Part 50, promulgated May 5, 1975 (40 FR 19439), provides numerical guidance for design objectives and technical specification requirements for limiting conditions of operation for light-water-cooled nuclear power plants.

The following regulatory guides are being prepared to assist in application of the numerical guidance in Appendix I:

1. Calculation of Annual Average Doses to Man from Routine Releases of Reactor Effluents for the Purpose of Implementing Appendix I.

2. Calculations of Releases of Radioactive Materials in Liquid and Gaseous Effluents from Pressurized Water Reactors (PWRs).

3. Calculation of Releases of Radioactive Materials in Liquid and Gaseous Effluents from Boiling Water Reactors (BWRs), and

4. Methods for Estimating Atmospheric Dispersion of Gaseous Effluents from Routine Releases.

region. More stringent design or effluent objectives or a larger exclusion area may be required in such cases.

While it is the concentration of radioactivity in the atmosphere at any distance from the point of release, χ (Ci/m³), that must be controlled, the ratio χ/Q , where Q (Ci/sec) is the rate of release of radioactivity from the source, has become a commonly evaluated term because it depends only on atmospheric variables and distance from the source.

If the atmospheric conditions are unfavorable with respect to dispersion characteristics at a proposed site, the exclusion area may have to be unusually large to satisfy the dose criteria of 10 CFR Part 100. If under assumed unfavorable atmospheric conditions (see Regulatory Guides 1.3 and 1.4) the dispersion of radioactivity released following a design basis accident is insufficient at the boundary of the exclusion area (see the following section, "Population Considerations") and the outer boundary of the low population zone, the site would not satisfy the requirements of 10 CFR Part 100. Thus, the design of the station would be required to include appropriate and adequate compensating engineered safety features.

Local fogging and icing can result from plumes discharged into the atmosphere from cooling towers, lakes, canals, or spray ponds, but can generally be acceptably mitigated by station design and operational practices. However, some sites have the potential for severe fogging or icing due to local atmospheric conditions. For example, areas of unusually high moisture content that are protected from large-scale airflow patterns are most likely to experience these conditions. The impacts are generally of greatest potential importance relative to transportation or electrical transmission corridors in the vicinity of a site.

A cooling system designed with special consideration for reducing drift may be required due to the sensitivity of the natural vegetation or the crops in the vicinity of the site to damage from airborne salt particles. The vulnerability of existing industries or other facilities in the vicinity of the site to corrosion by drift from cooling tower or spray system drift should be considered. Not only are the amount, direction, and distance of the drift from the cooling system important, but the salt concentration above the natural background salt deposition at the site is also important in assessing drift effects. None of these considerations are critical in evaluating the suitability of a site, but they could result in special cooling system design requirements or in the need for a larger site to confine the effects of drift within the site boundary. The environmental effects of salt drift are most severe where saline water or water with high mineral content is used for condenser cooling.

Cooling towers produce cloudlike plumes which vary in size and altitude depending on the atmospheric conditions. The plumes are often a few miles in length before becoming dissipated, but the plumes themselves or their shadows could have aesthetic impacts. Visible plumes emitted from cooling towers in the vicinity of airports could cause a hazard to aviation.

3. Population Considerations

A reactor licensee is required by 10 CFR Part 100 to designate an exclusion area and to have authority to determine all activities within that area, including removal of personnel and property. In selecting a site for a nuclear power station, it is necessary to provide for an exclusion area in which the applicant has such authority. The exclusion area must be of such size that doses to individuals at any point on its boundary for 2 hours immediately following the onset of a postulated fission product release are less than certain prescribed values. Transportation corridors, such as highways, railroads, and waterways, are permitted to traverse the exclusion area provided (1) these are not so close to the facility as to interfere with normal operation of the facility and (2) appropriate and effective arrangements are made to control traffic on the highway, railroad, or waterway in the case of emergency to protect the public health and safety.

As set forth in 10 CFR Part 100, a nuclear power station site must have a low population zone (LPZ) immediately surrounding the exclusion area in which the population is (a) sufficiently limited in number and (b) distributed in such a way that there is a reasonable probability that appropriate measures could be taken in their behalf in the event of a serious accident. A proposed site will also have a "population center distance," defined as the distance from the nuclear reactor to the nearest boundary of a densely populated center containing more than about 25 000 residents. The population center distance must be at least one and one-third times the distance to the outer boundary of the LPZ. However, 10 CFR Part 100 requires that the LPZ boundary be sufficiently remote that a release of fission products (calculated as a consequence of a postulated accident) will not result in radiation doses to individuals on the outer boundary of the LPZ greater than certain specified values.

WASH-1235, "The Site Population Factor, A Technique for Consideration of Population in Site Comparison," October 1974, discusses a methodology that is useful in comparing population distributions at alternative sites.

4. Hydrology

4.1 Flooding

Criteria for evaluation of seismically induced floods are provided in Appendix A to 10 CFR Part 100. Regulatory Guide 1.59 describes an acceptable method of determining the design basis floods for sites along streams or rivers and discusses the phenomena producing comparable design basis floods for coastal, estuary, and Great Lakes sites. The effects of a probable maximum flood (as defined in Regulatory Guide 1.59), seiche, surge, or seismically induced flood such as might be caused by dam failures or tsunami on station safety functions can generally be controlled by engineering design or protection of the safety-related structures, systems, and components which are identified in Regulatory Guide 1.29, "Seismic Design Classification." For some river valleys, flood plains, or areas along coastlines, there may not be sufficient information to make the evaluations needed to satisfy the criteria for seismically induced flooding. In such cases, study of the potential for dam failure, river blockage, or diversion in the river system or distantly and locally generated sea waves may be needed to determine the suitability of a site. In lieu of detailed investigations, Regulatory Guide 1.59 and Section 2.4 of Regulatory Guide 1.70 present acceptable analytical techniques for evaluating seismically induced flooding.

4.2 Water Availability

Nuclear power stations require reliable sources of water for steam condensation, service water, emergency core cooling system, and other functions. In regions where water is in short supply, the recirculation of the hot cooling water through cooling towers, artificial ponds, or impoundments has been practiced.

Essential water requirements for nuclear power plants are that sufficient water be available for cooling during plant operation and normal shutdown, for the ultimate heat sink,^a and for fire protection. The limitations imposed by existing laws or allocation policies govern the use and consumption of cooling water at potential sites^b for normal operation. Regulatory Guide 1.27

^aRegulatory Guide 1.27, "Ultimate Heat Sink for Nuclear Power Plants," provides guidance on water supply for the ultimate heat sink.

^bTo the extent that site selection is dependent on water diversions for consumptive use, allocation of water supply is a function of state statutory and administrative procedures.

A discussion of the establishment of state regulation of water use is provided in "Industrial Developments and the Environment, Legal Reforms to Improve the Decision-Making Process in Industrial Site Selection," Special Committee on Environmental Law of the American Bar Association, August 1973.

discusses the safety requirements. Consumptive use of water may necessitate an evaluation of existing and future water uses in the area to ensure adequate water supply during droughts both for station operation and other water users (i.e., nuclear power station requirements versus public water supply). Regulatory agencies should be consulted to avoid potential conflicts.

Where required by applicable law, demonstration of a request for certification of the rights to withdraw or consume water and an indication that the request is consistent with appropriate State and regional programs and policies should be provided as part of the application for a construction permit or operating license.

The availability of essential water during periods of low flow or low water level is an important initial consideration for identifying potential sites on rivers, small shallow lakes, or along coastlines. Both the frequency and duration of low flow or low level periods should be determined from the historical record and, if the cooling water is to be drawn from impoundments, from projected operating practices.

4.3 Water Quality

Thermal and chemical effluents discharged to navigable streams are governed by the Federal Water Pollution Control Act (FWPCA, PL 92-500), 40 CFR Part 122, 40 CFR Part 423, and State water quality standards. The applicant should also determine other regulations that are current at the time sites are under consideration. Section 401(a)(1) of the FWPCA requires, in part, that any applicant for an NRC construction permit for a nuclear power station provide to the NRC certification from the State that any discharge will comply with applicable effluent limitations and other water pollution control requirements. In the absence of such certification, no construction permit can be issued by the NRC unless the requirement is waived by the State or the State fails to act within a reasonable period of time. A National Pollution Discharge Elimination System (NPDES) permit to discharge effluents to navigable streams pursuant to Section 402 of the FWPCA may be required for a nuclear power station to operate in compliance with the Act, but is not a prerequisite to an NRC construction permit or operating license.

Evaluations of the dispersion and dilution capabilities and potential contamination pathways of the ground water environment under operating and accident conditions with respect to present and future users are required. Potential radiological and nonradiological contaminants of ground water should be evaluated. The

suitability of sites in areas with a complex ground water hydrology or of sites located over aquifers that are or may be used by large populations for domestic or industrial water supplies or for irrigation water can only be determined after reliable assessments have been made of the potential impacts of the reactor plants on the ground water.

Although management of the quality of surface waters is important, water quality per se is not a determining factor in assessing the suitability of a site since adequate design alternatives can generally be developed to meet the requirements of the Federal Water Pollution Control Act and the Commission's regulations implementing NEPA. However, the environmental characteristics or the complexity of the environment at a site and its vicinity may be such that it would be difficult to obtain or develop sufficient information to establish, in a timely manner, that the potential environmental impacts on water quality would be acceptable. Examples of situations that could pose unusual impact assessment or design problems are areas of existing marginal water quality, small bays, estuaries, stratified waters, and sites that would require intake from and discharge to waters of markedly different quality, such as intake of marine water and discharge to an estuary.

The following are examples of potential environmental effects of station construction and operation that must be assessed: physical and chemical environmental alterations in habitats of important species including plant-induced rapid changes in environmental conditions; changes in normal current direction or velocity of the cooling water source and receiving water; scouring and siltation resulting from construction and cooling water intake and discharge; alterations resulting from dredging and spoil disposal; and interference with shoreline processes.

5. Ecological Systems and Biota

Areas of great importance to the local aquatic ecosystem may present major difficulties in assessing potential impacts on populations of important species or ecological systems. Such areas include those used for breeding (e.g., nesting and spawning), wintering, and feeding, as well as areas where there may be seasonally high concentrations of individuals of important species.² Where the ecological sensitivity of a site under consideration cannot be established from existing information, more detailed studies, as discussed in Regulatory

Guide 4.2, may be necessary. Impacts of station construction² and operation on the biota and ecological systems may be mitigated by design and operational practices if justifiable relative to costs and benefits. In general, the important considerations in the balancing of costs and benefits are (a) the uniqueness of a habitat or ecological system within the region under consideration and (b) the amount of habitat or ecological system that would be destroyed or disrupted relative to the total amount of the habitat or ecological system present in the region or the vulnerability of the reproductive capacity of important species populations to the effects of construction and operation of the plant and ancillary facilities.

The alteration of one or more of the existing environmental conditions may render a habitat unsuitable as a breeding or nursery area. In some cases, organisms use identical breeding and nursery areas each year; if the characteristics of the areas are changed, breeding success may be substantially reduced or enhanced. Destruction of part or all of a breeding or nursery area may cause population shifts that result in increased competition for the remaining suitable areas. Such population shifts cannot compensate for the reduced size of the breeding or nursery areas if the remaining suitable area is already occupied by the species. Some species will desert a breeding area because of man's activities in the proximity to the area, even in the absence of physical disturbance of the actual breeding area.

Of special concern relative to site selection are those unique or especially rich feeding areas that might be destroyed, degraded, or made inaccessible to important species by station construction or operation. Evaluation

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more of the following criteria applies:

- (1) If the species is commercially or recreationally valuable,
- (2) If the species is endangered or threatened,
- (3) If the species affects the well-being of some important species within criteria (1) or (2) or if it is critical to the structure and function of a valuable ecological system or is a biological indicator of radionuclides in the environment.

Endangered and threatened species are defined by PL 93-205, the Endangered Species Act of 1973, as follows: "The term 'endangered species' means any species which is in danger of extinction throughout all or a significant portion of its range other than a species of the Class Insecta determined by the Secretary to constitute a pest whose protection under the provisions of this Act would present an overwhelming and overriding risk to man." "The term 'threatened species' means any species which is likely to become an endangered species within the foreseeable future throughout all or a significant portion of its range." Lists of endangered and threatened species are published periodically in the *Federal Register* by the Secretary of the Interior.

²A compilation of construction practices is provided in "General Environmental Guidelines for Evaluating and Reporting the Effects of Nuclear Power Plant Site Preparation, Plant and Transmission Facilities Construction," Atomic Industrial Forum, February 1974.

²A species, whether animal or plant, is important (for the purpose of this guide) if a specific causal link can be identified between the nuclear power station and the species and if one or

(Continued)

of feeding areas in relation to potential construction or operation impacts includes the following considerations: size of the feeding area onsite in relation to the total feeding area offsite, food density, time of use, location in relation to other habitats, topography relative to access routes, and other factors (including man's activities). Site modification may reduce the quality of feeding areas by destruction of a portion of the food base, destruction of cover, or both.

Construction and operation of nuclear power stations can create barriers to migration, occurring mainly in the aquatic environment. Narrow zones of passage for migratory animals in some rivers and estuaries may be restricted or blocked by station operation. Partial or complete blockage of a zone of passage may result from the discharge of heat or chemicals to receiving water bodies or the construction and placement of power station structures in the water body. Strong-swimming aquatic animals often avoid waters of adverse quality, but larval and immature forms are usually moved and dispersed by water currents. It is therefore important in site selection that the routes and times of movement of the immature stages be considered in relation to potential effects.

A detailed assessment of potential impact on the species population would be required for sites where placement of intake or discharge structures would markedly disrupt normal current patterns in migration paths of important species. The potentials for impingement of organisms on cooling water intake structures and entrainment of organisms through the cooling system are determined by a number of variables including site characteristics, intake structure design, and placement of the structures at the site.

Site characteristics should be considered relative to design and placement of cooling system features and the potential of the cooling system to hold fish in an area longer than the normal period of migration or to entrap resident populations in areas where they would be adverse. Affected, either directly or indirectly, by limited food supply or adverse temperatures. Canals or areas where cooling waters are discharged may induce fish to remain in an unnaturally warmed habitat. The cessation of station operation during winter can be lethal to these fish because of an abrupt drop in water temperature.

6. Land Use and Aesthetics

Many impacts on land use at the site and in the site neighborhood due to construction and operation of the plant, transmission lines, and transportation corridors can be mitigated by appropriate designs and practices. Aesthetic impacts can be reduced by selecting sites where existing topography and forests can be utilized for

screening station structures from nearby scenic, historical, or recreational resources. Restoration of natural vegetation, creative landscaping,^b and the integration of structures with the environment can mitigate adverse visual impacts.

Preconstruction archeological excavations can usually reduce losses. Short-term salvage archeology may not be sufficient if extensive or valuable archeological sites are found on the potential site for a nuclear station. For areas of archeological concern, the Chief Archeologist of the National Park Service is an information source, as are the State Archeologist and the State Liaison Officer responsible for the National Historic Preservation Act activities for a particular state.

Proposed alternative land use may render a site unsuitable for a nuclear power station. For example, lands specified by a community (1) as planned for other uses or (2) as restricted to compatible uses vis-a-vis other lands may be unsuitable. Therefore, official land use plans developed by governments at any level and by regional agencies should be consulted for possible conflicts with power station siting. A list of Federal agencies that have jurisdiction or expertise in land use planning, regulation, or management has been published by the Council on Environmental Quality.^a

Another class of impacts involves the preempting of existing land use at the site itself. For example, nuclear power station siting in areas uniquely suited for growing specialty crops may be considered a type of land conversion involving unacceptable economic dislocation.

Sites adjacent to lands devoted to public use may be considered unsuitable. In particular, the use of some sites or transmission line or transportation corridors close to special areas administered by Federal, State, or local agencies for scenic or recreational use may cause unacceptable impacts regardless of design parameters. Such cases are most apt to arise in areas adjacent to natural-resource oriented areas (e.g., Yellowstone National Park) as opposed to recreation-oriented areas (e.g., Lake Mead National Recreation Area). Some historical and archeological sites may also fall into this category. The acceptability of sites near special areas of public use should be determined by consulting cognizant government agencies.

^bStation protection requirements for nuclear safeguards may influence landscape design and clearing of vegetation.

^aSee U.S. Council on Environmental Quality, "Preparation of Environmental Impact Statements: Guidelines," 38 FR 20549, August 1, 1973.

The following Federal agencies should be consulted for the special areas listed:

a. National Park Service (U.S. Department of the Interior)

National Parks; International Parks; National Memorial Parks; National Battlefields, Battlefield Parks and Battlefield Sites; National Military Parks; Historic Areas and National Historic Sites; National Capital Parks; National Monuments and Cemeteries; National Seashores and Lakeshores; National Rivers and Scenic Riverways; National Recreation Areas; National Scenic Trails and Scientific Reserves; National Parkways

b. National Park Service Preservation Program

National Landmarks Program; Historic American Buildings Survey; National Register of Historic Places; National Historical Landmarks Program; National Park Service Archeological Program

c. Bureau of Sport Fisheries and Wildlife (U.S. Department of Interior)

National Wildlife Refuges

d. Forest Service (U.S. Department of Agriculture)

National Forest Wilderness, Primitive Areas, National Forests.

Individual States and local governments administer parks, recreation areas, and other public use and benefit areas. Information on these areas should be obtained from cognizant State agencies such as State departments of natural resources. (See publications such as the "Conservation Directory 1973: A Listing of Organizations, Agencies and Officials Concerned with Natural Resource Use and Management," published by the National Wildlife Federation for state-by-state references.) The Advisory Council on Historic Preservation or the appropriate State historical society should be contacted for information on historic areas.

It should be recognized that some areas, as yet undesignated, may be unsuitable for siting because of public interest in future dedication to public scenic, recreational, or cultural use. Relatively rare land types such as sand dunes and wetlands are prime candidates for such future designation. However, the acceptability of sites for nuclear power stations at some future time in these areas will depend on the existing impacts from industrial, commercial, and other developments.

7. Industrial, Military, and Transportation Facilities

Potential accidents at present or projected nearby in-

dustrial, military, and transportation facilities may affect the safety of a nuclear power station.² A site should not be selected if, in the event of such an accident, it is not possible to safely shut down a plant at that site or if it is not possible to have nearby facilities alter their mode of operation or incorporate features to reduce to an acceptable level the likelihood and severity of such potential accidents.

In the event of an accident at a nearby industrial facility such as a chemical plant, refinery, mining and quarrying operation, oil or gas well, or gas and petroleum product storage installation, it is possible that missiles, shock waves, flammable vapor clouds, toxic chemicals, or incendiary fragments may result. These may affect the station itself or the station operators in a way that jeopardizes the safety of the station.

Regulatory Guide 1.78, "Assumptions for Evaluating the Habitability of a Nuclear Power Plant Control Room During a Postulated Hazardous Chemical Release," describes assumptions acceptable to the NRC staff for use in assessing the habitability of the control room during and after a postulated external release of hazardous chemicals and describes criteria that are generally acceptable to the staff for the protection of the control room operators.

Nearby military facilities, such as munitions storage areas and ordnance test ranges, may threaten station safety. The acceptability of a site depends on establishing, among other things, that the nuclear power station can be designed so its safety will not be affected by an accident at the military installation. Alternatively, an otherwise unacceptable site may become acceptable if the cognizant military organization agrees to change the installation or mode of operation to reduce the likelihood or severity of potential accidents involving the nuclear station to an acceptable level.

An accident during the transport of hazardous materials (e.g., by air, waterway, railroad, highway, or pipeline) near a nuclear power plant may generate shock waves, missiles, and toxic or corrosive gases which can affect the safe operation of the station. The consequences of the accident will depend on the proximity of the transportation facility to the site, the nature and maximum quantity of the hazardous material per shipment, and the layout of the nuclear station. Unless the station can be designed to operate safely in the event of a postulated accident or an enforceable agreement can be reached to limit the transport of hazardous materials or the transportation link can be relocated, the proposed site may not be acceptable.

²Section 2.2 of Regulatory Guide 1.70 lists these safety considerations.

Airports are transportation facilities that pose specialized hazards to nearby nuclear power stations. Potential threats to stations from aircraft result from the aircraft itself as a missile and from the secondary effects of a crash, e.g., fire.

8. Socioeconomics

Social and economic issues are important determinants of siting policy. It is difficult both to assess the nature of the impacts involved and to determine value schemes for predicting the level or the acceptability of potential impacts.

The siting, construction, and operation of a nuclear power station may have significant impacts on the socioeconomic structure of a community and may place severe stresses on the local labor supply, transportation facilities, and community services in general. There may be changes in the tax basis and in community expenditures, and problems may occur in determining equitable levels of compensation for persons relocated as a result of the station siting. It is usually possible to resolve such difficulties by proper coordination with impacted communities; however, some impacts may be locally unacceptable and too costly to avoid by any reasonable program for their mitigation. Evaluation of the suitability of a site should therefore include consideration of purpose and probable adequacy of socioeconomic impact mitigation plans for such economic impacts on any community where local acceptance problems can be reasonably foreseen.

Certain communities in a site neighborhood may be subject to unusual impacts that would be excessively costly to mitigate. Among such communities are towns that possess a notably distinctive cultural character, i.e., towns that have preserved or restored numerous places of historic interest, have specialized in an unusual industry or avocational activity, or have otherwise markedly distinguished themselves from other communities.

9. Noise

Noise levels at nuclear stations occur during both the construction and operation phases and could have unacceptable impacts. Cooling towers, turbines, and transformers contribute to the noise levels during station operation.

C. REGULATORY POSITION

1. Geology/Seismology

Sites that include capable faults, as defined in Appendix A to 10 CFR Part 100, are not suitable for nuclear

power stations. The state of the art has not progressed to the point at which it is possible to design a nuclear power station for surface or near-surface displacement with a sufficiently high level of confidence to ensure that the integrity of the safety-related features of the plant will remain intact.

Sites within about 5 miles of a surface capable fault greater than 1,000 feet in length are usually not suitable for a nuclear power station. In any case, extensive and detailed geologic and seismic field studies and analyses should be conducted for such a proposed site.

Sites located near geologic structures for which an adequate data base to determine "capability" does not exist at the time of application are likely to be subject to a longer licensing process in view of the need for extensive and detailed geologic and seismic investigations of the site and surrounding region and for the rigorous analyses of the site-plant combination.

Sites with competent bedrock for foundations generally have suitable foundation conditions. In regions where there are few or no such sites, it is prudent to select sites in areas with competent and stable solid soils, such as dense sands and glacial tills. Other materials may also provide satisfactory foundation conditions, but in any case, a detailed geologic and geotechnical investigation will be required to determine static and dynamic engineering properties of the material underlying the site in accordance with Sections IV(a)(4) and V(d) of Appendix A to 10 CFR Part 100.

2. Atmospheric Extremes and Dispersion

As noted in Section B.2 of this guide, site atmospheric conditions are site suitability characteristics principally with respect to the calculation of radiation doses resulting from the release of fission products as a consequence of a postulated accident and the establishment of exclusion area boundary, low population zone boundary, and distance to a population center. Accordingly, the regulatory position on atmospheric dispersion of radiological effluents is incorporated into the following section, "Population Considerations."

Nonradiological atmospheric considerations such as local fogging and icing, cooling tower drift, cooling tower plume lengths and plume interactions between cooling tower plumes, and plumes from nearby industrial facilities should be considered in evaluating the suitability of potential sites.

3. Population Considerations

Areas of low population density are preferred for nuclear power station sites. High population densities projected for any time during the lifetime of a station are considered during both the NRC staff review and the public hearing phases of the licensing process. If the population density at the proposed site is not acceptably low, then the applicant will be required to give special attention to alternative sites with lower population densities.

If the population density, including weighted transient population, projected at the time of initial operation of a nuclear power station exceeds 500 persons per square mile averaged over any radial distance out to 30 miles (cumulative population at a distance divided by the area at that distance), or the projected population density over the lifetime of the facility exceeds 1,000 persons per square mile averaged over any radial distance out to 30 miles, special attention should be given to the consideration of alternative sites with lower population densities.

Transient population should be included for those sites where a significant number of people (other than those just passing through the area) work, reside part-time, or engage in recreational activities and are not permanent residents of the area. The transient population should be taken into account by weighting the transient population according to the fraction of time the transients are in the area.

Based on past experience, the NRC staff has found that a minimum exclusion distance of 0.4 mile, even with unfavorable design basis atmospheric dispersion characteristics, usually provides assurance that engineered safety features can be designed to bring the calculated dose from a postulated accident within the guidelines of 10 CFR Part 100. If the minimum exclusion distance is less than 0.4 mile, it may be necessary to place special conditions on the station design (e.g., added engineered safety features) before the requirements of 10 CFR Part 100 are met. Also, based on past experience, the staff has found that a distance of 3 miles to the outer boundary of the low population zone is usually adequate.

4. Hydrology

4.1 Flooding

To evaluate sites located in river valleys, on flood plains, or along coastlines where there is a potential for flooding, the site suitability studies described in Regula-

tory Guide 1.59, "Design Basis Floods for Nuclear Power Plants," should be made.

4.2 Water Availability

A highly dependable system of water supply sources must be shown to be available under postulated occurrences of natural and site-related accidental phenomena or combinations of such phenomena as discussed in Regulatory Guide 1.59.

To evaluate the suitability of sites, there should be reasonable assurance that permits for consumptive use of water in the quantities needed for a nuclear power plant of the stated approximate capacity and type of cooling system can be obtained by the applicant from the appropriate State, local, or regional bodies.

4.3 Water Quality

The potential impacts of nuclear power stations on water quality are likely to be acceptable if effluent limitations, water quality criteria for receiving waters, and other requirements promulgated pursuant to the Federal Water Pollution Control Act are applicable and satisfied.

The criteria provided in 10 CFR Parts 20 and 50 will be used by the NRC staff for determining permissible concentrations of radioactive materials discharged to surface water or to ground water.⁴

Aquifers that are or may be used by large populations for domestic, municipal, industrial, or irrigation water supplies provide potential pathways for the transport of radioactive material to man in the event of an accident. To evaluate the suitability of proposed sites located over such aquifers, detailed studies of factors identified in Section 2.4.13 of Regulatory Guide 1.70, "Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants," should be completed.

5. Ecological Systems and Biota

The ecological systems and biota at potential sites and their environs should be sufficiently well known to allow reasonably certain predictions that there would be no unacceptable or unnecessary deleterious impacts on populations of important species or on ecological systems with which they are associated from the construction or operation of a nuclear power station at the site.

⁴Appendix I to 10 CFR Part 50 provides numerical guidance for design objectives and technical specification requirements for limiting conditions of operation for light-water-cooled nuclear power stations.

When early site inspections and evaluations indicate that critical or exceptionally complex ecological systems will have to be studied in detail to determine the appropriate plant designs, proposals to use such sites should be deferred unless sites with less complex characteristics are not available.

It should be determined whether any important species (as defined in Section B.5 of this guide) inhabit or use the proposed site or its environs; and the relative abundance and distribution of their populations should be considered. Potential adverse impacts on important species should be identified and assessed. The relative abundance of individuals of an important species inhabiting a potential site should be compared to available information in the literature concerning the total estimated local population. Any predicted impacts on the species should be evaluated relative to effects on the local population and the total population of the species. The destruction of, or sublethal effects on, a number of individuals which would not adversely affect the reproductive capacity and vitality of a population or the crop of an economically important harvestable population or recreationally important population should generally be acceptable, except in the case of certain endangered species. If there are endangered or threatened species at a site, the potential effects should be evaluated relative to the impact on the local population and the total estimated population over the entire range of the species as noted in the literature.

It should be determined whether there are any important ecological systems at a site or in its environs. If so, determination should be made as to whether the ecological systems are especially vulnerable to change or if they contain important species habitats, such as breeding areas (e.g., nesting and spawning areas), nursery, feeding, resting, and wintering areas, or other areas of seasonally high concentrations of individuals of important species.

The important considerations in the balancing of costs and benefits include the following: the uniqueness of a habitat or ecological system within the region under consideration, the amount of the habitat or ecological system destroyed or disrupted relative to the total amount in the region, and the vulnerability of the reproductive capacity of important species populations to the effects of construction and operation of the station and ancillary facilities.

If sites contain, are adjacent to, or may impact on important ecological systems or habitats that are unique, limited in extent, or necessary to the productivity of populations of important species (e.g., wetlands and estuaries), they cannot be evaluated as to suitability for a nuclear power station until adequate assessments for the reliable prediction of impacts have been completed

and the facility design characteristics that would satisfactorily mitigate the potential ecological impacts have been defined. In areas where reliable and sufficient data are not available, the collection and evaluation of appropriate seasonal data may be required.

Migrations of important species and migration routes that pass through the site or its environs should be identified. Generally, the most critical migratory routes relative to nuclear power station siting are those of aquatic species in water bodies associated with the cooling systems. Site conditions that should be identified and evaluated in assessing potential impacts on important aquatic migratory species include (1) narrow zones of passage, (2) migration periods that are coincident with maximum ambient temperatures, (3) potential for major modification of currents by station structures, (4) potential for increased turbidity during construction, and (5) potential for entrapment, entrainment, or impingement by or in the cooling water system, or blocking of migration by facility structures or effluents.

The potential blockage of movements of important terrestrial animal populations due to the use of the site for a nuclear power station and the availability of alternative routes that would provide for maintenance of the species' breeding population should be assessed.

If justifiable relative to costs and benefits, potential impacts of plant construction and operation on the biota and ecological systems can generally be mitigated by adequate engineering design and site planning and by proper construction and operation practice when there is adequate information about the vulnerability of the important species and ecological systems.

A summary of environmental considerations, parameters, and regulatory positions for use in evaluating the suitability of sites for nuclear power stations is provided in Appendix B to this guide. A discussion of ecological systems and habitats, the level of detail that should be addressed in the site selection process, and the survey, monitoring, and analytical techniques for assessing impacts on important species and ecological systems will be summarized in subsequent appendices to this guide.

6. Land Use and Aesthetics

Land use plans adopted by Federal, State, regional, or local governmental entities should be examined, and any conflict between these plans and use of a potential site should be resolved by consultation with the appropriate governmental entity.

For potential sites on land devoted to specialty crop production where changes in land use might result in market dislocations, a detailed investigation should be

provided to demonstrate that potential problems have been identified and resolved.

The potential aesthetic impact of nuclear power stations at sites near natural-resource oriented public use areas is of particular concern, and evaluation of the suitability of such sites is dependent on consideration of specific station design layout. However, existing aesthetic impacts at potential sites should be taken into account as mitigating any requirements for further special design.

7. Industrial, Military, and Transportation Facilities

Potentially hazardous facilities and activities within 5 miles of a proposed site should be identified. If a preliminary evaluation of potential accidents at these facilities indicates that the potential hazards from shock waves and missiles approach or exceed those of the design basis tornado for the region² or potential hazards such as flammable vapor clouds, toxic chemicals, or incendiary fragments exist, the suitability of the site should be determined by detailed evaluation of the degree of risk imposed by the potential hazard.

The identification of design basis events resulting from the presence of hazardous materials or activities in the vicinity of a nuclear power station is acceptable if the design basis events include each postulated type of accident for which a realistic estimate of the probability of occurrence of potential exposures in excess of the 10 CFR Part 100 guidelines exceeds approximately 10^{-7} per year. Because of the difficulty of assigning precise numerical values to the probability of occurrence of the types of potential hazards generally considered in determining the acceptability of sites for nuclear stations, judgment must be used as to the acceptability of the overall risk presented by an event.

In view of the low probability events under consideration, the probability of occurrence of the initiating events leading to potential consequences in excess of 10 CFR Part 100 exposure guidelines should be based on assumptions that are as realistic as is practicable. In addition, because of the low probability events under consideration, valid statistical data are often not available to permit accurate quantitative calculation of probabilities. Accordingly, a conservative calculation showing that the probability of occurrence of potential exposures in excess of the 10 CFR Part 100 guidelines is approximately 10^{-6} per year is acceptable if, when combined with reasonable qualitative arguments, the realistic probability can be shown to be lower.

The effects of design basis events have been appropriately considered if analyses of the effects of those accidents on the safety-related features of the proposed nuclear station have been performed and appropriate measures (e.g., hardening, fire protection) to mitigate the consequences of such events have been taken.

To evaluate the suitability of sites in detail for potential accidents involving hazardous materials and activities at nearby industrial, military, and transportation facilities, the studies described in Section 2.2 of Regulatory Guide 1.70 should be made.

8. Socioeconomics

The NRC staff considers that an evaluation of the suitability of nuclear power station sites near distinctive communities should demonstrate that the construction and operation of the nuclear station, including transmission and transportation corridors, and potential problems relating to community services, such as schools, police and fire protection, water and sewage, and health facilities, will not adversely affect the distinctive character of the community. A preliminary investigation should be made to identify and analyze problems that may arise due to the proximity of a distinctive community to a proposed site.

9. Noise

Noise levels at proposed sites must comply with applicable Federal, State, and local noise regulations.

D. IMPLEMENTATION

The purpose of this section is to provide information to applicants and licensees regarding the NRC staff's plans for using this regulatory guide.

Since this guide reflects current NRC staff practice with regard to the implementation of existing regulations concerning site suitability, it can be used immediately to indicate considerations that should be addressed in the initial stage of the site selection process to identify potential sites for nuclear power stations.

²The design basis tornado is described in Regulatory Guide 1.76, "Design Basis Floods for Nuclear Power Plants."

APPENDIX A**SAFETY-RELATED SITE CONSIDERATIONS
FOR ASSESSING SITE SUITABILITY
FOR NUCLEAR POWER STATIONS**

This appendix provides a checklist of safety-related site characteristics, relevant regulations and regulatory guides, and regulatory experience and position for assessing site suitability for nuclear power stations.

Considerations	Relevant Regulations and Regulatory Guides	Regulatory Experience and Position
A.1 Geology/Seismology Geologic and seismic characteristics of a site, such as surface faulting, ground motion, and foundation conditions (including liquefaction, subsidence, and landslide potential), may affect the safety of a nuclear power station.	10 CFR Part 100, Appendix A, "Seismic and Geologic Siting Criteria for Nuclear Power Plants." Regulatory Guide 1.70, Chapter 2 (identifies safety-related site characteristics). Regulatory Guide 1.29 (discusses plant safety features which should be controlled by engineering design).	Sites that include capable faults are not suitable for a nuclear power station. Sites within about 5 miles of a surface capable fault (greater than 1000 feet in length) are generally not suitable for a nuclear power station. Sites should be selected in areas for which an adequate geologic data base exists to determine "capability." Delay in licensing can result from a need for extensive geologic and seismic investigations. Conservative design of safety-related structures will be required when geologic, seismic, and foundation information is questionable. Sites with competent bedrock generally have suitable foundation conditions. If bedrock sites are not available, it is prudent to select site in areas known to have a low subsidence and liquefaction potential. Investigations will be required to determine the static and dynamic engineering properties of the material underlying the site as stated in 10 CFR Part 100, Sec. IV(a)(4) and Sec. V(d) of Appendix A.

Considerations	Relevant Regulations and Regulatory Guides	Regulatory Experience and Position
A.2 Atmospheric Dispersion The atmospheric conditions at a site should provide sufficient dispersion of radioactive materials released during a postulated accident to reduce the radiation exposures of individuals at the exclusion area and low population zone boundaries to the values prescribed in 10 CFR Part 100.	10 CFR Part 100, "Reactor Site Criteria." Regulatory Guide 1.23, "Onsite Meteorological Programs." Regulatory Guide 1.3, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss of Coolant Accident for Boiling Water Reactors." Regulatory Guide 1.4, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss of Coolant Accident for Pressurized Water Reactors." Regulatory Guide 1.5, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Steam Line Break Accident for Boiling Water Reactors." Regulatory Guide 1.24, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Pressurized Water Reactor Radioactive Gas Storage Tank Failure." Regulatory Guide 1.25, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Fuel Handling Accident in the Fuel Handling and Storage Facility for Boiling and Pressurized Water Reactors."	Unfavorable safety-related design basis atmospheric dispersion characteristics can be compensated for by an adequate exclusion distance and engineered safety features. Accordingly, the regulatory position on atmospheric dispersion of radiological effluents is incorporated into the section "Population Considerations" (see A.3 of this appendix).

Considerations	Relevant Regulations and Regulatory Guides	Regulatory Experience and Position
A.3 Population Considerations In the event of a serious accident at a nuclear power station, effective action must be taken to minimize exposure of individuals outside the station to any radioactive materials which may be released during the accident. To ensure that exposure to populations will be minimized in the event of an accident, the nuclear power station should not be located in a densely populated area.	10 CFR Part 100, "Reactor Site Criteria," requires the following: • An "exclusion area" surrounding the reactor in which the reactor licensee has the authority to determine all activities, including exclusion or removal of personnel and property; • A "low population zone" (LPZ) which immediately surrounds the exclusion area in which the population number and distribution is such that "there is a reasonable probability that appropriate measures could be taken in their behalf in the event of a serious accident;" • At any point on the exclusion area boundary and on the outer boundary of the LPZ the exposure of individuals to a postulated release of fission products (as a consequence of an accident) be less than certain prescribed values. • That the "population center distance," defined as the distance from the nuclear reactor to the nearest boundary of a densely populated center having more than 25,000 residents, be at least one and one-third the distance from the reactor to the outer boundary of the LPZ. Regulatory Guides 1.3, 1.4, 1.5, 1.24, and 1.25 give calculational methods (see A.2 of this appendix.)	If the population density, including weighted transient population, projected at the time of initial operation of a nuclear power station exceeds 500 persons per square mile averaged over any radial distance out to 30 miles (cumulative population at a distance divided by the area at that distance), or the projected population density over the lifetime of the facility exceeds 1,000 persons per square mile averaged over any radial distance out to 30 miles, special attention should be given to the consideration of alternative sites with the lower population densities. Transient population should be included for those sites where a significant number of people (other than those just passing through the area) work, reside part-time, or engage in recreational activities, and are not permanent residents of the area. The transient population should be taken into account by weighting the transient population according to the fraction of time the transients are in the area. Based on past experience, the NRC staff has found that a minimum exclusion distance of 0.4 mile,^a even with the most unfavorable design basis atmospheric dispersion characteristics, provides assurance that engineered safety features can be added that will bring the calculated doses from a postulated accident within the guidelines of 10 CFR Part 100. If the minimum exclusion distance is less than 0.4 mile, it may be necessary to place special conditions on station design (e.g., added engineered safety features) before the site can be considered acceptable. Also based on past experience, the NRC staff has found that a distance of 3 miles to the outer boundary of the LPZ is usually adequate.^a

^a The guidelines numbers for exclusion area and LPZ are based on historical siting experience of light-water-cooled reactors. In certain instances different dimensions have been established for high temperature gas-cooled reactors.

Considerations	Relevant Regulations and Regulatory Guides	Regulatory Experience and Position
A.4 Hydrology A.4.1 Flooding Precipitation, wind, or seismically induced flooding (e.g., resulting from dam failure, from river blockage or diversion, or from distantly and locally generated sea waves) can affect the safety of a nuclear power station.	10 CFR Part 100, Appendix A, "Seismic and Geologic Siting Criteria for Nuclear Power Plants." Regulatory Guide 1.59, "Design Basis Floods for Nuclear Power Plants." Regulatory Guide 1.70, "Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants," (Section 2.4). 10 CFR Part 50, Appendix A, "General Design Criteria for Nuclear Power Plants," Criterion 2, "Design Bases for Protection Against Natural Phenomena."	To evaluate sites located in river valleys, on flood plains, or along coastlines where there is a potential for flooding, the studies described in Regulatory Guide 1.59 should be made.
A.4.2 Water Supply A safety-related water supply is required for normal or emergency shutdown and cooldown.	10 CFR Part 100, Appendix A, "Seismic and Geologic Siting Criteria for Nuclear Power Plants." Regulatory Guide 1.59, "Design Basis Floods for Nuclear Power Plants." Regulatory Guide 1.27, "Ultimate Heat Sink for Nuclear Power Plants."	A highly dependable system of water supply sources should be shown to be available under postulated occurrences of natural phenomena and site-related accidental phenomena or combinations of such phenomena as discussed in Regulatory Guide 1.59. To evaluate the suitability of a site, there must be reasonable assurance that permits for water use and for water consumption in the quantities needed for a nuclear power plant of the stated approximate capacity and type of cooling system can be obtained by the applicant from the appropriate State, local, or regional bodies.
A.4.3 Water Quality Contamination of ground water and surface water by radioactive materials discharged from nuclear stations could cause public health hazards.	10 CFR Part 20, "Standards For Protection Against Radiation" 10 CFR Part 50, "Licensing of Production and Utilization Facilities."	The criteria provided in 10 CFR Parts 20 and 50 will be used by the NRC staff for determining permissible concentrations of radionuclides discharged to surface water and ground water.

Considerations	Relevant Regulations and Regulatory Guides	Regulatory Experience and Position
A.5 Industrial, Military, and Transportation Facilities Near the Site Accidents at present or projected nearby industrial, military, and transportation facilities may affect the safety of the nuclear power station.	10 CFR Part 50, Appendix A, "General Design Criteria for Nuclear Power Plants," Criterion 4, "Environmental and Missile Design Bases." Regulatory Guide 1.70, "Standard Format and Content of Safety Analysis Reports," Section 2.2 (lists types of facilities and potential accidents). Regulatory Guide 1.78, "Assumptions for Evaluating the Habitability of a Nuclear Power Plant Control Room During a Postulated Hazardous Chemical Release."	Potentially hazardous facilities and activities within miles of a proposed site must be identified. If a preliminary evaluation of potential accidents of these facilities indicates that the potential hazards from shock waves and missiles approach or exceed those of the design basis tornado for the region (the design basis tornado is described in Regulatory Guide 1.76), or potential hazards such as flammable vapor clouds, toxic chemicals, or incendiary fragments exist, the suitability of the site should be determined by detailed evaluation of the potential hazard. The identification of design basis events resulting from the presence of nearby hazardous materials or activities in the vicinity of a nuclear power station is acceptable if the design basis events include each postulated type of accident for which a realistic estimate of the probability of occurrence of potential exposures in excess of 10 CFR Part 100 guidelines exceeds approximately 10^{-7} per year. To evaluate the suitability of sites in detail for potential accident situations involving hazardous materials and activities from nearby industrial, military, and transportation facilities, the studies described in Section 2.2 of Regulatory Guide 1.70 should be made.

APPENDIX B

**ENVIRONMENTAL CONSIDERATIONS FOR ASSESSING
SITE SUITABILITY FOR NUCLEAR POWER STATIONS**

This appendix summarizes environmental considerations related to site characteristics that should be addressed in the early site selection process. The relative importance of the different factors to be considered varies with the region or State in which the potential sites are located.

Site selection processes can be facilitated by establishing limits for various parameters based on the best judgment of specialists knowledgeable of the region under consideration. For example, limits can be chosen for the

fraction of water that can be diverted in certain situations without adversely affecting the local populations of important species. Although simplistic because important factors such as the distribution of important species in the water body are not taken into account, such limits can be useful in a screening process for site selection.

A discussion of performance characteristics of light-water-cooled reactor stations which may affect the environment is given in WASH-1355, "Nuclear Power Facility Performance Characteristics for Making Environmental Impact Assessments," December 1974.

Considerations	Parameters	Regulatory Position
B.1 Preservation of Important Habitats Important habitats are those that are essential to maintaining the reproductive capacity and vitality of important species populations² or the harvestable crop of economically or recreationally important species. Such habitats include breeding areas (e.g., nesting and spawning areas), nursery, feeding, resting, and wintering areas or other areas of seasonally high concentrations of individuals of important species. The construction and operation of nuclear power stations (including new transmission lines and access corridors constructed in conjunction with the station) can result in the destruction or alteration of habitats of important species leading to changes in the abundance of a species or in the species composition of a community.	The proportion of an important habitat that would be destroyed or significantly altered in relation to the total habitat within the region in which the proposed site is to be located is a useful parameter for estimating potential impacts of the construction or operation of a nuclear power station. The value of the proportion varies among species and among habitats. The region considered in determining proportions is the normal geographic range of the specific population in question. If endangered or threatened species occur at a site, the potential effects of the construction and operation of a nuclear power station should be evaluated relative to the potential impact on the local population and the total estimated population over the entire range of the species. See also Chapter 2 of Regulatory Guide 4.2, "Preparation of Environmental Reports for Nuclear Power Stations."	In general, a detailed justification should be provided when the destruction or significant alteration of more than a few percent of important habitat types is proposed. The reproductive capacity of populations of important species and the harvestable crop of economically or recreationally important populations must be maintained unless justification for proposed or probable changes can be provided.

²As defined for this guide in Section B.

Considerations	Parameters	Regulatory Position
B.2 Migratory Routes of Important Species Seasonal or daily migrations are essential to maintaining the reproductive capacity of some important species populations. Disruption of migratory patterns can result from partial or complete blockage of migratory routes by structures, discharge plumes, environmental alterations, or man's activities (e.g., transportation or transmission corridor clearing and site preparation).	The width or cross-sectional area of a water body at a proposed site relative to the general width or cross-sectional area in the portion of the water body used by migrating species should be estimated. Suggested minimum zones of passage range from 1/3 to 3/4 of the width or cross-sectional areas of narrow water bodies.^{a,b} Some species migrate in central, deeper areas while others use marginal, shallow areas. Rivers, streams, and estuaries are seldom homogeneous in their lateral dimension with respect to depth, current velocity, and habitat type. Thus, the use of width or cross-sectional area criteria for determining adequate zones of passage should be combined with a knowledge of important species and their migratory requirements.	Narrow reaches of water bodies should usually be avoided as sites for locating intake or discharge structures. A zone of passage that will permit normal movement of important species populations and maintenance of the harvestable crop of economically important populations should be provided.

^a *Water Quality Criteria*, 1972, National Academy of Sciences - National Academy of Engineering, Washington, D.C., 1972.

^b *Handbook of Environmental Control, Volume III: Water Supply and Treatment*, R.G. Bond and C.P. Straub (Editors), CRS Press, Cleveland, Ohio, 1973.

Considerations	Parameters	Regulatory Position
B.3 Entrainment and Impingement of Aquatic Organisms	The depth of the water body at the point of intake relative to the general depth of the water body in the vicinity of the site.	The site should have characteristics that allow placement of intake structures where the relative abundance of important species is small and where low approach velocities can be attained. (Deep regions are generally less productive than shallow areas. It is not implied that benthic intakes are necessary.)
The reproductive capacity of important species populations may be impaired by lethal stresses or by sublethal stresses that affect reproduction of individuals or result in increased predation on the affected species population.	The proportion of water withdrawn relative to the net new available water at the site is an indirect measure of the destruction of plankton which in turn is indicative of possible effects on populations of important species. It has been suggested that the fraction of available new water that can be diverted is in the range of 10% to 20% of flow.^{b,c}	Important habitats (see B.1) should be avoided as locations for intake structures.
Fish and other aquatic organisms can be killed or injured by impingement on cooling water intake screens^a or by entrainment in discharge plumes.	This simplistic parameter (proportion of water withdrawal) is suitable for use in a screening process for site selection. However, other factors such as distribution of important species should be considered and in all cases the advice of experts on the local fisheries should be consulted to ensure that proposed withdrawals will not be excessive.	

^aApproach velocity and screen-face velocity are design criteria that may affect the impingement of larger organisms, principally fish, on intake screens. Acceptable approach and screen-face velocities are based on fish swim speeds which will vary with the species, site, and season.

^b*The Water's Edge: Critical Problems of the Coastal Zone*, B.H. Ketchum (Editor), MIT Press, Cambridge, Mass., 1972.

^c*Engineering for Resolution of the Energy-Environment Dilemma*, National Academy of Engineering, Washington, D.C., 1971.

Considerations	Parameters	Regulatory Position
B.4 Entrapment of Aquatic Organisms Cooling water intake and discharge system features, such as canals and thermal plumes, can attract and entrap organisms, principally fish. The resulting concentration of important fish species near the station site can result in higher mortalities from station-related causes, such as impingement, cold shock, or gas bubble disease, than would otherwise occur. Entrapment can also interrupt normal migratory patterns.	Site characteristics that will accommodate design features that mitigate or prevent entrapment.	Sites where the construction of intake or discharge canals would be necessary should be avoided unless the site and important species characteristics are such that entry of important species to the canal can be prevented or limited by screening.

Considerations	Parameters	Regulatory Position
B.5 Water Quality Effluents discharged from nuclear power plants are governed under the authority of the Federal Water Pollution Control Act (FWPCA) (PL 92-500).	Applicable EPA-approved State water quality standards. For states without EPA-approved water quality standards, the water quality criteria listed in <i>Water Quality Criteria, 1972</i>²³ will be used for evaluation.	Pursuant to Section 401(a)(1) of the FWPCA, certification from the State that any discharge will comply with applicable effluent limitations and other water pollution control requirements is necessary before the NRC can issue a construction permit unless the requirement is waived by the State or the State fails to act within a reasonable length of time. Issuance of a permit pursuant to Section 402 of the Act is not a prerequisite to an NRC license or permit. Where station construction or operation has the potential to degrade water quality to the possible detriment of other users, more detailed analyses and evaluation of water quality may be necessary.

Considerations	Parameters	Regulatory Position
B.6 Water Availability The consumptive use of water for cooling may be restricted by statute, may be inconsistent with water use planning, or may lead to an unacceptable impact to the water resource.	Applicable Federal, State, and local statutory requirements. Compatibility with water use plan of cognizant water resource planning agency. In the absence of a water use plan, the effect on other water users is evaluated considering flow or volume reduction and the resultant ability of all users to obtain adequate supply and to meet applicable water quality standards (see B.5, Water Quality).	Water use and consumption must comply with statutory requirements and be compatible with water use plans of cognizant water resources planning agencies. Consumptive use should be restricted such that the supply of other users is not impaired and that applicable surface water quality standards could be met, assuming normal station operational discharges and extreme low flow conditions defined by generally accepted engineering practices. For multipurpose impounded lakes and reservoirs, consumptive use should be restricted such that the magnitude and frequency of draw-down will not result in unacceptable damage to important habitats (see B.1, Preservation of Important Habitats) or be inconsistent with the management goals for the water body.

Considerations	Parameters	Regulatory Position
B.7 Established Public Amenity Areas Areas dedicated by Federal, State, or local governments to scenic, recreational, or cultural purposes are generally prohibited areas for siting power stations. Siting nuclear power stations in the vicinity of established public amenity areas could result in the loss or deterioration of important public amenities.	Proximity to public amenity area. Viewability (see B.10, Visual Amenities).	Siting in the vicinity of designated public amenity areas will generally require extensive evaluation and justification. The evaluation of the suitability of sites in the vicinity of public amenity areas is dependent on consideration of a specific plant design and station layout in relation to potential impacts on the public amenity area.
B.8 Prospective Designated Amenity Areas Areas containing important resources for scenic, recreational, or cultural use may not currently be designated as such by public agencies but may involve a net loss to the public if converted to power generation. These areas may include locally rare land types, such as sand dunes, wetlands, or coastal cliffs.	Comparison of possible amenity areas in number and extent with other similar areas available on a local, regional, or national basis, as appropriate.	Public amenity areas that are distinctive, unique, or rare in a region should be avoided as sites for nuclear power stations

Considerations	Parameters	Regulatory Position
B.11 Local Fogging and Icing Water and water vapor released to the atmosphere from recirculating cooling systems can lead to ground fog and ice resulting in transportation hazards and damage to electric transmission systems.	Increase in number of hours of fogging or icing caused by operation of the station.	The hazards on transportation routes from fog or ice that result from station operation should be evaluated. The evaluation should include estimates of frequency of occurrence of station-induced fogging and icing and their impact on transportation, electrical transmission, and other activities and functions.
B.12 Cooling Tower Drift Concentrations of chemicals, dissolved solids, and suspended solids in cooling tower drift could affect terrestrial biota and result in unacceptable damage to vegetation and other resources.	The percent drift loss from recirculating condenser cooling water, particle size distribution, salt deposition rate, local atmospheric conditions, and loss of sensitive terrestrial biota affected by salt deposition from cooling tower drift.	The potential loss of important terrestrial species and other resources should be considered.

Considerations	Parameters	Regulatory Position
B.13 Cooling Tower Plume Lengths Natural draft cooling towers produce cloudlike plumes which vary in size and altitude depending on the atmospheric conditions. The plumes are usually a few miles in length before becoming dissipated, although plume lengths of 20 to 30 miles have been reported from cooling towers. Visible plumes emitted from cooling towers could cause a hazard to commercial and military aviation in the vicinity of commercial and military airports. The plumes themselves or their shadows could have aesthetic impacts.	The number of hours per year the plume is visible as a function of direction and distance from the cooling towers.	The visibility of cooling tower plumes as a function of direction and distance from cooling towers should be considered. The evaluation should include estimates of frequency of occurrence for plumes as well as potential hazards to aviation in the vicinity of commercial and military airports.
B.14 Plume Interaction Water vapor from cooling tower plumes may interact with industrial emissions from nearby facilities to form noxious or toxic substances which could cause adverse public health impacts, or result in unacceptable levels of damage to biota, structures, and other resources.	The degree to which impacts may occur will vary depending on the distance between the nuclear and fossil-fueled sites, the hours per year of plume interaction, the type and concentration of chemical reaction products, the area of chemical fallout, and the local atmospheric conditions.	The hazards to public health, structures, and other resources from potential plume interaction between cooling tower plumes and plumes from fossil-fueled sites and industrial emissions from nearby facilities should be considered.

APPENDIX D

GENERAL ADMINISTRATIVE ACTIVITIES

Wm. B. Cottrell*

[Nucl. Safety, 15(6): 757-759 (November-December 1974)]

The topical reviews in this article summarize some significant safety developments, not covered elsewhere, during July and August 1974. The Advisory Committee on Reactor Safeguards (ACRS) is making public the minutes of its meetings pertaining to siting evaluations, and the AEC has extended the time by which existing reactors would have to meet the new emergency core-cooling system (ECCS) criteria. Civil penalties have been imposed, or are proposed, for three licensees. The Directorate of Licensing has issued guidance on the use of "standardized" plants to expedite licensing. Some seven Atomic Safety and Licensing Board (ASLB) and/or Atomic Safety and Licensing Appeal Board (ALAB) rulings of special interest have been excerpted. Nine new (or revised) regulatory guides were published, and some 20 reports of safety interest were issued by various groups. Finally, miscellaneous safety-related items with appropriate references are mentioned.

ACRS DOCUMENTS RELATING TO SITING EVALUATION

The AEC has announced¹ that it will make public most of the remaining substantive internal documents relating to the activities of the ACRS since it was founded in 1957. The ACRS is an independent group established by law to advise the AEC on reactor safety matters.

The ACRS has previously released more than 25,000 pages of the minutes of its meetings from 1965 to 1973 as part of an agreement between the AEC and intervenors in three nuclear power-plant licensing proceedings and a request from Ralph Nader. Additional thousands of pages of documents of the AEC Regulatory Staff, its consultants, and national laboratories previously were provided as part of these proceedings.

The ACRS believes that its latest action will provide the public with a more complete understanding of the nature and scope of its activities than has been provided to date by the limited release of documents in response to specific court orders or to requests under the Freedom of Information Act.² All such documents will be placed in the AEC Public Document Room at 1717 H Street, NW, Washington, D. C. The complete text of the Aug. 13, 1974, letter on the subject "ACRS Documents Relating to Siting Evaluation" follows:³

In connection with recent matters related to the Freedom of Information Act, the ACRS has been requested to provide minutes of its meetings concerning the potential siting of power reactors at sites having greater surrounding population densities than sites previously approved. These minutes are being made available by the ACRS with minimum deletion. It should be noted that prior to the implementation of the FACA, the Committee's minutes were intended only for the internal use of the Committee and were not necessarily reviewed or certified as correct as now required by that Act. Accordingly, they should in general be regarded as informal reports meant to reflect the flavor and principal features of the meetings, rather than as detailed, or completely accurate accountings of the proceedings.

For those who read these documents, the following explanation may be helpful.

The Committee is concerned primarily with the avoidance of discharges of radioactive materials from nuclear power reactors under conditions which conceivably could lead to undue risks to public health and welfare.

To aid it in its work, the Committee retains a large group of consultants encompassing many disciplines. It is also aided by the reports and work reviews of the Regulatory Staff, the designers of the nuclear steam supply systems, the nuclear power plant architect-engineers, and a large number of research and development groups.

We have already accumulated in the United States about two hundred reactor years of experience at commercial nuclear power stations without a major radioactive discharge or other nuclear related accident that has resulted in injury to the public. Since there have been no such accidents in the past, the probability of such accidents

* Oak Ridge National Laboratory.

occurring in the future cannot be arrived at by any analysis of this experience. However, the Committee has from the beginning assessed, and continues to assess, the potential for and the possible consequences of postulated events. In pursuing this aspect of safety evaluation, the Committee thus continues to deal with highly unlikely but potentially very serious accidents. Examples of this type of accident are pressure vessel failure, a complete loss of power, or an uncontained loss of coolant accident. One aspect of such considerations is the comparison of the effects of the discharge of arbitrary quantities of radioactive material at different sites. This type of analysis was employed in a site comparison study of the Metropolitan Siting Subcommittee (subsequently called the Siting Evaluation Subcommittee).

The Siting Evaluation Subcommittee attempted to develop information which would aid the full Committee in the formulation of a policy on the siting of large power reactors at sites having greater surrounding populations than those previously approved. The need for such a policy had become apparent as a result of some actual applications and potential proposals involving sites having population characteristics somewhat "worse" than those for the most populous sites already approved.

The activities of the Subcommittee and of the full Committee toward this objective involved many meetings and discussions within the Committee and with the Regulatory Staff, the Commission, and representatives of the utilities and the nuclear industry. These activities contributed to the development of the position that large power reactors might be acceptable at sites having population densities somewhat greater than those previously approved, but only if certain kinds of improvements were made in their design and construction and if additional safeguards were provided. This position was expressed in various drafts of reports (letters) which were prepared in a formal manner in order to coalesce the views of the Committee into a coherent collegial position. This basic position was made public in the Committee's letter of September 10, 1969, relating to the proposed Newbold Island site.

In the course of the Subcommittee activity discussed above, two things became evident. First, a quantitative basis for comparing sites in terms of population distribution and density was needed, in order to compare one site with another in respect to population and to identify those sites to which the proposed criteria would apply. And second, the criteria of 10 CFR Part 100 would not in themselves provide a quantitative basis for assessing the acceptability of more densely populated sites, since the quantitative limits of these criteria could be satisfied by providing additional engineered safety features in lieu of distance as a means of protecting the public against the consequences of the design basis accidents required to be considered.

The criteria of 10 CFR Part 100, together with other rules, criteria and guides promulgated by the AEC, provide a basis for the design of a large power reactor that would not present an undue hazard to the health and safety of the public at even a heavily populated site in the event of any of the postulated low-probability design basis accidents. However, an accident much worse than those considered and having a lower probability of occurrence (Class 9)

could present a threat to the public which would require additional consideration of the population density surrounding the site.

For these reasons, the Subcommittee's approach to the development of a siting index to be used in identifying "worse" sites involved assumptions of arbitrary, large releases of radioactivity. This approach was explored as a possible means of development of a usable method for evaluating quantitatively, on a comparative basis, the population densities (and distributions) associated with reactor sites. The objective was to achieve an agreed-upon formula which utilizes the population data at a site to produce a single-number index (Site Population Index), the magnitude of which indicates whether the site is "better" or "worse" than another site (in respect to population), and by how much.

The objective described was not fully achieved. However, after prolonged effort there was developed a possible approach to a Site Population Index which the Committee thought could be used as an initial point of departure in a joint ACRS-Regulatory Staff effort directed toward eventual development of an Index suitable for public release and for use by both industry and the Regulatory bodies.

The principal general approach taken in the attempt to develop an Index was as follows. A series of hypothetical release, of large magnitude is postulated for the site under consideration, and the same series is postulated for a "reference" site. This reference site, as eventually conceived, is assumed to have a population density which is at every point equal to the arithmetic mean of the population densities at the corresponding points of the Indian Point and Zion sites, these being the most highly populated sites employed to date. The effects of the assumed releases on the population at the considered site are then calculated on a simplistic but a consistent basis and compared to those similarly calculated for the reference site, the comparison being expressed in the form of a single-number Index.

The effects calculated are two: the man-rem exposures resulting from the releases; and, the number of people who would have to be evacuated or would otherwise be subjected to a dose exceeding some specified threshold value. The calculation of these effects is based on an assumed rate of attenuation of dose with distance from the reactor; usually dose is assumed as inversely proportional to distance to the three halves power.

For each of the postulated releases, the total "exposure" (man-rem) is found by the above process for the site under study, and for the reference site. The ratio between these two numbers is averaged over all releases considered. Likewise, for each of these releases, the number of hypothetical "evacuees" (or persons receiving threshold doses) is found for the site under study and for the reference site, and the ratio of these two numbers is averaged over all releases postulated. The two averages are then combined to give a single-number Index. The Index is unity for the reference site, less than 1 for a "better" (less populated) site, and greater than 1 for a "worse" site.

In the Index model last considered (and presented to the Regulatory Staff in 1970 for possible further development into a useful regulatory tool), the smallest release considered was arbitrarily assumed to cause doses of 25 R

at 0.5 miles, and the largest to cause doses of 25 R at 6 miles. An arbitrary plume width was assumed. The lower cut-off dose for man-rem calculation was about 1 R. The specified threshold dose value was taken as 25 R.

In some earlier Index models explored, the releases hypothesized were arbitrarily assumed to cause doses twenty times higher than those above, and the specified threshold dose value was taken to be 500 R, also twenty times higher. Not surprisingly, with these arbitrarily high assumed releases and with a further arbitrary assumption of *no evacuation* of any portion of the populace, at highly populated sites very high man-rem exposures resulted and very high numbers of people were calculated to receive hypothetical doses of 500 R or more, i.e., became "fatalities", as indicated in certain of the minutes. It is emphasized that these man-rem figures and "fatality" numbers do not result from analyses of specific or realistic accident

sequences. It should also be noted that associated estimates of the probability of occurrence of such consequences were not made except for general cognizance that such a probability is exceedingly low, and for the larger of the hypothetical consequences, probably decades below that of a Part 100 type release.

The exploration by the Committee of models which resulted in such numbers was done for purposes of comparison of one site with another (with identical calculations made at both sites), and not as a means of appraising any site on an absolute basis.

The Committee believes that a site population index of the general type treated represents a useful concept, and continues to employ informally the developmental index already formulated as an aid in judging on a comparative basis the population magnitudes at sites it is called upon to appraise.

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* Prepared by Shirley P. Hendrix.

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