PRM-50-121

From:	Parillo, John
To:	RulemakingComments Resource
Subject:	Re-Submittal - 10 CFR 2.802 Petition for rulemaking Accident Dose Criteria
Date:	Saturday, November 23, 2019 4:32:30 PM
Attachments:	Petition for Rule Making Accident Dose Criteria Volume I Main Text and Appendicies for Submittal.docx Petition for Rule Making Accident Dose Criteria Volume II References .docx

Secretary, Attention: Rulemakings and Adjudications Staff

Pursuant to 10 CFR 2.802, please find attached a petition for rulemaking. The proposed rule would allow licensees to adopt revised accident dose acceptance criteria as an alternative to the accident dose criteria specified in § 50.67 Accident source term. The revised accident dose criteria would be described in a separate voluntary rule § 50.67(a) specifying a uniform value of 100 milli Sieverts (10 rem) for the off-site locations and for the control room.

Problem Description:

The U.S. Nuclear Regulatory Commission's (NRC's) design basis accident (DBA) dose criteria and the resulting design of accident mitigation systems could be perceived to emphasize protection of the control room operator over protection of the public. The control room criterion restricts the calculated 30-day accident dose to the annual occupational limit of five rem while the off-site dose criteria allows for a calculated dose of 25 rem in two hours. The off-site dose criteria were derived from the siting practices of the earliest reactors and are not reflective of current health physics knowledge or modern plant construction. As a result, the design of accident mitigation systems may not be optimized in the best interest of NRC's mission of protecting public health and safety. The control room accident dose criterion has proven to be challenging to demonstrate with most plants having very little margin to the regulation.

Proposed Solution:

The proposed voluntary rule would allow licensees to adopt revised accident dose criteria that will; (1) be reflective of modern health physics recommendations and modern plant designs, (2) provide a better balance between protection of the control room operator and protection of the public, and (3) relieve the unnecessary regulatory burden associated with meeting the current control room dose criterion.

The attached petition includes the history of the current dose criteria, proposed changes to § 50.67 Accident source term and General Deign Criterion 19, corresponding revisions to Regulatory Guide 1.183, Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors, as well as other supporting information.

The petitioner has attempted to gain support from NRC staff to initiate rulemaking through internal processes for over ten years without success. The referenced dose criteria are codified in NRC regulations. Since internal processes such as the Non-Concurrence Process and the Differing Profession Opinion process are not applicable to concerns with regulations, the petitioner reluctantly submits the attached § 2.802 Petition for rulemaking as an individual.

The attached petition has been reviewed to ensure that refenced materials are publicly

available

Respectfully,

John G. Parillo

Attachments:

Petition for Rulemaking - Accident Dose Criteria

Volume I, Main Text and Appendices

Appendix A: The Influence of Early Reactor Siting on DBA Acceptance Criteria

Appendix B: Proposed Revisions to § 50.67 Accident source term and GDC-19

Appendix C: Proposed Revision to RG 1.183 Accident Dose Criteria

Appendix D: Comparisons of Footnotes Describing the 25 rem Dose Criterion in Current
Regulations to the Original Footnote in 10 CFR 100.11

Appendix E: MCA (LOCA) Doses for Operating Plants

Volume II, References

1. Advisory Committee on Reactor Safeguards, Minutes of the Environmental Subcommittee, February 18, 1959, ML021750385.

2. Atomic Energy Commission, 10 CFR Chapter 1, Power and Test Reactors, Notice of Proposed Rule Making [on Reactor Site Criteria], (24 FRN 4184 1959), May 23, 1959.

3. Advisory Committee on Reactor Safeguards, Minutes of the Environmental Subcommittee, August 23, 1960, ML021750500.

4. Atomic Energy Commission, Report to General Manager by the Director, Division of Licensing & Regulations, Reactor Site Criteria. This report contains an important ACRS letter dated October 22, 1960. This important letter can be found on pages 21 -25 of Reference 4. The entire report is available in Adams with a Document Date May 25, 1959, ML021960199.

5. Atomic Energy Commission, 10 CFR Part 100 Reactor Site Criteria, Notice of Proposed Guides, (26 FRN 1224 1961), February 11, 1961.

6. Atomic Energy Commission, Title 10 Atomic Energy, Chapter I, Atomic Energy Commission, Part 100, Reactor Site Criteria, (27 FRN 3509 1962), April 13, 1962.

7. Relevant FRN excerpts discussing the conversion of the §100.11 criteria to 25 rem TEDE.

8. Raymond A Crandall, Petition for Rulemaking to U.S. NRC, PRM-50-87, to revise 10

CFR 50, Appendix A, "General Design Criteria for Nuclear Power Plants" and 10 CFR 50.67, "Accident Source Term," to eliminate Control Room dose criteria. Docketed May 25, 2007.

9. U.S. nuclear Regulatory Commission, 10 CFR Part 50, NRC-2007-0016; PRM-50-97, Raymond A. Crandall; Denial of Petition for Rulemaking, Federal Register Vol. 7415 January 26, 2009.

10. Health Effects Associated with Radiation Exposure. This reference is a compilation of the health effects associated with exposure to radiation from the Environmental Protection Agency, The Centers for Disease Control and Prevention and the Radiation Safety Division of Duke University and Duke Medicine.

11. Excerpts from, A Brief History of Radiation - Protection Standards, William C. Inkret, Charles B. Meinhold, and John C. Taschner, Los Alamos Science, Number 23 1995. Available at https://permalink.lanl.gov › object › lareport › LA-UR-95-4005-04. This reference shows how recommendations for safe levels of radiation exposure have been reduced as scientists learned more about the health effects of radiation.

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PETITION FOR RULEMAKING
DESIGN BASIS ACCIDENT DOSE CRITERIA

PURPOSE:

The U.S. Nuclear Regulatory Commission's (NRC's) design basis accident (DBA) dose criteria and the resulting design of accident mitigation systems could be perceived to emphasize protection of the control room operator over protection of the public. The control room criterion restricts the calculated 30-day accident dose to the annual occupational limit of five rem while the off-site dose criteria allows for a calculated dose of 25 rem in two hours. DBA dose criteria should not be viewed as representing actual doses received by individuals but rather as figures of merit which have a direct impact on the design of structures, systems and components (SSCs) important to safety. The off-site dose criteria were derived from the siting practices of the earliest reactors and are not reflective of current health physics knowledge or modern plant construction. As a result, the design of accident mitigation systems may not be optimized in the best interest of NRC's mission of protecting public health and safety. The control room accident dose criterion has proven to be challenging to demonstrate with many plants having very little margin to the regulation.

The purpose of this petition is to identify concerns with current DBA dose criteria and to recommend a proposed voluntary rule allowing licensees to adopt revised accident dose acceptance criteria that will; (1) be reflective of modern health physics recommendations and modern plant designs, (2) provide a better balance between protection of the control room operator and protection of the public, and (3) relieve the unnecessary regulatory burden associated with meeting the current control room dose criterion.

SUMMARY:

During the 1950s, applicants for reactor construction permits submitted Hazards Summary Reports to the Atomic Energy Commission (AEC) describing the potential dose consequences from what was considered the "maximum credible accident."[1] These evaluations contained wide variations in both the assumed source terms as well as the proposed dose acceptance criteria. In response to the recognition that more definitive siting criteria was needed, the AEC developed a procedural methodology to define reactor siting criteria that was generally consistent with the siting practices in effect at the time. There was a concern within the AEC that it was premature to codify these criteria so early in the development of the nuclear power industry. Notwithstanding this concern, in 1962, the AEC published 10 CFR Part 100, "Reactor Site Criteria", specifying dose acceptance criteria of 25 rem whole body and 300 rem thyroid for a 2 hour period at the Exclusion Area Boundary (EAB) and for the accident duration at the outer boundary of the Low Population Zone (LPZ).

The stated objective of the reactor siting criteria was to avoid serious injury to individuals if an unlikely, but still credible, accident should occur. Both the 25 rem criterion and the concept of an exclusion area addressed the potential for extreme radiological hazards that would exist if a

[1] The maximum credible accident (also referred to as the maximum probable or maximum hypothetical accident) is that accident whose consequences, as measured by the radiation exposure of the surrounding public, would not be exceeded by any other accident whose occurrence during the lifetime of the facility would appear to be credible. Such accidents have generally been assumed to result in substantial meltdown of the core with subsequent release into containment of appreciable quantities of fission products. These evaluations assume containment integrity with offsite hazards evaluated based on design basis containment leakage.

fuel melt source term was released into an unshielded containment[2]. The regulation states that the 25 rem whole body corresponds to the once-in-a-lifetime accidental or emergency dose for radiation workers which according to 1959 national council on radiation protection (NCRP) recommendations may be disregarded in the determination of their radiation exposure status[3]. There is no analogous citation for the 300 rem thyroid dose criterion which was not the dose equivalent to 25 rem whole body. Radiation protection standards at the time would have suggested a 6:1 ratio of thyroid to whole body dose (resulting in 150 rem) so the 300 rem was somewhat arbitrary. The codification of site criteria fulfilled the need to reduce the subjective nature of judging site suitability while providing a methodology that did not conflict with siting decisions already made by the AEC. The regulation was intended to be an interim measure until the state-of-the-art allowed for more definitive standards to be developed.

In 1971 Appendix A, "General Design Criteria for Nuclear Power Plants," was added to 10 CFR Part 50. General Design Criterion 19 (GDC-19) specified that adequate protection shall be provided to permit access and occupancy of the control room for the duration of an accident without exceeding a radiation exposure of 5 rem whole body or its equivalent to any part of the body. The originally stated objective for the 5 rem control room accident dose criterion is not readably traceable however the NRC staff believes that the primary objective of the criterion was to provide a safe, comfortable environment that would enable the control room operators to focus attention on accident mitigation. The numerical value chosen fulfilled this objective however the alignment of the control room accident dose criterion with the annual limit for occupational dose has been an ongoing challenge for licensees. The 5 rem control room dose criterion is limiting for many licensees and this raises the question regarding whether a slightly higher value could still satisfy the objective of providing a comfortable environment for the operators while reducing regulatory burden by increasing the small margin many licensees have relative to the current acceptance criterion.

In the late 1970s there were concerns within the NRC that siting practices were not providing enough emphasis on site isolation as an important contributor to defense-in-depth because engineered safety feature (ESF) systems could be designed to make almost any site acceptable from an accident dose calculation point of view. In August 1978, the NRC directed the staff to develop a general policy statement on nuclear power reactor siting which resulted in NUREG-0625, "Report of the Siting Policy Task Force," recommending that fixed distances should be required for the EAB and the LPZ in lieu of dose consequence analyses. After numerous comments objecting to a proposed rule (57 FR 47802), which was based on NUREG-0625 recommendations, the commission decided to retain source term and dose calculations by relocating a new single dose criterion based on total effective dose equivalent (TEDE) in 10 CFR 50.34 (61 FR 65157 December 11, 1996).

The new TEDE criterion is applicable to all new reactors and existing reactors that choose to adopt the alternative source term (AST) methodology. Depending on the contribution to TEDE dose from iodine in the released source term, the 25 rem TEDE criterion allows for the associated thyroid dose to substantially exceed the previously controlling 300 rem thyroid limitation. Therefore, new reactors are being sited with a less restrictive dose criterion than the earliest reactors.

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[2] Spherical steel containment structures approximately one inch thick were common at the time and offered a minimal degree of radiation shielding. Appendix A, Volume I of this petition provides more detail on the influence of early reactor siting on DBA acceptance criteria.

[3] NRC regulations do not allow any exposures to be disregarded for radiation exposure status purposes; all exposures are counted.

Modern health physics recommendations suggest that a dose of 25 rem is difficult to justify as adequately fulfilling the objective of not causing serious harm especially when considering the most dose-sensitive members of the public. The same health physics recommendations indicate that the 5 rem control room dose criterion may be overly restrictive.

Therefore, it is recommended that a uniform design basis accident dose criterion of 10 rem TEDE for the control room, EAB, and LPZ boundary be available to licensees on a voluntary basis. Adoption of this voluntary rule would result in a less restrictive control room dose criterion while significantly strengthening the offsite dose criterion. This voluntary change would provide various benefits in that: (1) it is technically defensible based on modern health physics guidance indicating that an increased cancer risk is not expected for exposures below 10 rem; (2) it would avoid the poor optics of allowing a higher design basis dose criterion for members of the public (including the most dose-sensitive groups such as children and pregnant women) than for highly trained nuclear professionals occupying the control room; (3) it would motivate licensees to provide greater emphasis on offsite dose reduction commensurate with NRC's mission to protect public health and safety; and (4) it would reduce the regulatory burden required to demonstrate the unnecessarily restrictive 5 rem control room dose criterion.

A significant number of plants would be able to meet a uniform 10 rem TEDE dose criterion without making any changes to their dose consequence analyses. Those plants whose existing DBA dose analyses would be challenged by a 10 rem TEDE dose criterion may be able to increase the credit taken for mitigation systems designed to limit releases to the environment while achieving an increased margin in their control room dose analyses. However, no action on the part of any licensees would be required since the proposed rule presented herein would be available for adoption on a voluntary basis.

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SCOPE OF PETITION:

This petition identifies concerns resulting from an examination of the dose criteria described in Title 10 of the *Code of Federal Regulations* (10 CFR) Part 100, "Reactor Site Criteria," as stated in 10 CFR 100.11, "Determination of exclusion area, low population zone, and population center distance," and its basis document, Technical Information Document (TID)-14844, "Calculation of Distance Factors for Power and Test Reactor Sites," USAEC, March 23, 1962. Additionally, this petition examines the objectives of the control room dose criteria in 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities," Appendix A, "General Design Criteria for Nuclear Power Plants," Criterion 19, "Control room," and the relationship between the control room criteria and the reactor site criteria. This petition also identifies concerns with the translation of the 10 CFR 100.11 dose criteria (25 rem whole body and 300 rem thyroid) into the single total effective dose equivalent (TEDE) criterion (25 rem TEDE) which is applicable to: applicants for a construction permit applying on or after January 10, 1997, in accordance with 10 CFR 50.34; all applicants under 10 CFR Part 52; and, existing plants originally licensed prior to January 10, 1997, that choose to adopt the alternative source term (AST) under 10 CFR 50.67, "Accident source term." This petition examines current recommendations from the health physics community that lend strong credibility to the thesis that the current NRC accident dose design criterion for the control room may be unnecessarily low while the criterion for members of the public may not properly define adequate protection. Finally, this petition recommends a proposed voluntary rule to better align the objectives of ensuring protection of the control room operator while maintaining adequate protection of the public.

DISCUSSION:

Hazard Summary Reports issued in the 1950's included the dose consequences from a maximum credible accident (MCA) also referred to as a maximum hypothetical accident (MHA) or a maximum probable accident (MPA). Such evaluations were based on the assumption that the plant experienced a substantial core melt releasing appreciable quantities of fission products into the containment atmosphere. These evaluations assumed containment integrity with offsite hazards evaluated based on design basis containment leakage. Applicants then evaluated the off-site radiological conditions for such an event and proffered various suggestions for dose acceptance criteria. The AEC evaluated these applications on a case by case basis without the benefit of a prescribed set of assumptions regarding the degree of core damage or defined dose acceptance criteria. There was a considerable effort in the AEC and the advisory committee on reactor safeguards (ACRS) during the time from 1958 through 1962 to devise a more systematic method to evaluate the licensee's MCA determinations. These concerns were described in an AEC report to the General Manager[4] by the Director of Licensing and Regulation on Reactor Site Criteria[5] as shown below:

> "The hazards reports as presented by the various applicants have shown a wide variation in estimating the magnitude of the maximum credible accident and in the dose

[4] The position of general manager appears to have been somewhat equivalent to the position of the executive director for operations in today's NRC as described in the following quote, "Five Commissioners appointed by the President would exercise authority for the operation of the Commission, while a general manager, also appointed by the President, would serve as chief executive officer." The Atomic Energy Commission," Alice Buck, U.S. department of Energy, 1983.

[5] Atomic Energy Commission, Reactor Site Criteria, Report to the General Manager by the Director, Division of Licensing and Regulation, Document Date May 25, 1959. This report is included in its entirety as Reference 4 of Volume II of this petition and can be accessed in Agencywide Documents and Management System (ADAMS), Accession No. ML021960199.

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

The promulgation of 10 CFR Part 100 and its basis document TID-14844 served to reduce the amount of subjectivity involved to the evaluation of reactor site suitability by defining the degree of core damage to be assumed in the MCA and by prescribing dose acceptance criteria.

Formally Stated Objectives of 10 CFR Part 100

The AEC first published a Notice of Proposed Rule Making regarding site criteria in 1959 (24 Federal Register Notice (FRN) 4184 1959)[6] announcing that:

> "The Commission is considering the formulation of an amendment to its regulations to state site criteria for the evaluation of proposed sites for nuclear power and test reactors and is publishing for comment safety factors which might be a basis for the development of site criteria."
>
> "In view of the complex nature of the environment, the wide variation in environmental conditions from one location to another and the variations in reactor characteristics and associated protection which can be engineered into a reactor facility, definitive criteria for general application to the siting problems have not been set forth."

The FRN went on to describe in general terms the need to show that, "the occurrence of any credible accident, will not create undue hazard to the health and safety of the public." The FRN described the general concept of an exclusion area under the complete control of the licensee as well as an area of low population density immediately outside the exclusion area.

In 1961, the AEC published 10 CFR Part 100, Reactor Site Criteria, Notice of Proposed Guides, (26 FRN 1224 1961)[7]. These guides were more descriptive and included specific dose criteria as well as an appendix detailing an example calculation of reactor siting distances. This FRN also included a more definitive set of objectives stating that:

> "The basic objectives which it is believed can be achieved under the criteria set forth in the proposed guides, are:
>
> > (a) Serious injury to individuals off-site should be avoided if an unlikely, but still credible, accident should occur;

[6] 24 FRN 4184 1959, May 23, 1959 is included in its entirety as Reference 2, Volume II of this petition.

[7] 26 FRN 1224 1961, February 11, 1961 is included in its entirety as Reference 5, Volume II of this petition.

> (b) Even if a more serious accident (not normally considered credible) should occur, the number of people killed should not be catastrophic;
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> (c) The exposure of large numbers of people in terms of total population dose should be low. The Commission intends to give further study to this problem in an effort to develop more specific guides on this subject. Meanwhile, in order to give recognition to this concept the population center distances to very large cites may have to be greater than those suggested by these guides."

There were numerous comments[8] received on the proposed Part 100 Site Criteria published for comment on February 11, 1961. There was general agreement that the proposed site criteria represented a distinct improvement over the criteria published on May 23, 1959. There was a concern over the inclusion of the Appendix which was felt to be too descriptive to include in a rule. In addition, there were several comments that objected to the wording of the objectives especially in paragraph (b), "Even if a more serious accident (not normally considered credible) should occur, the number of people killed should not be catastrophic."

The objectives stated in the proposed guides published on February 11, 1961 were not repeated in the final rule which was published on April 13, 1962. The final rule (27 FRN 3509 1962)[9] included the following discussion concerning the objective of the population center distance described in 10 CFR Part 100:

> "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. 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 specification of such a distance requirement 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 some cases where very large cities are involved, the population center distance may have to be greater than those suggested by these guides."

Background on the Development of 10 CFR Part 100 – Reactor Site Criteria

In order to gain a better understanding of the objectives of 10 CFR Part 100, it is instructive to examine some of the discussions that occurred during the development of the Reactor Siting Criteria. The minutes from an ACRS Environmental Subcommittee meeting held on February

[8] A detailed compilation of the comments received as of July 25, 1961 on the proposed rule published on February 11, 1961 was documented and is available in ADAMS Accession No. ML021750298.

[9] 27 FRN 3509 1962 April 13, 1962 is included in its entirety as Reference 6, Volume II of this petition.

19, 1959[10], included a discussion of issues related to site criteria. The minutes included the following discussion of the concept of an acceptable emergency dose for an accident condition:

> "Dr. McCullough differentiated normal vs. abnormal operation. In the course of normal operations one can expect cladding failures, stuffing box leaks, bearing failures, etc. These may result in release of some radioactivity and Part 20[11] should govern. Part 20 does not however apply to the abnormal operation (accident) brought about by cracking of a pump casing, rupture of high pressure piping, etc. It is not possible therefore to define an acceptable emergency dose since one cannot predict the accident. The concept of 25 R as an acceptable emergency dose is not valid. It is valid, of course, under the concept for which it was initially defined."
>
> "This was in connection with the willingness to expose an individual to a dose, which could be fairly accurately estimated in advance, in order to save life or valuable property. Further, one should have interest beyond the exposure to an individual at the site boundary. What doses are seen as a function of distance beyond the site boundary and how many people are exposed?"

The concept of an exclusion Area as it related to concerns for direct gamma shine from unshielded containments under accident conditions was included in the minutes from the February 19, 1959 ACRS meeting as follows:

> "There was considerable discussion about the necessary exclusion area around reactors of different powers. Although it is obvious that the selection of certain arbitrary distances for reactors of various powers may be a simple solution, some thought should be given to the basic reason for an exclusion area."
>
> "It has been generally stated that exclusion area is for the purpose of protecting against gamma shine in case of accidents and also to give a certain amount of time for warning, evacuation, or other alleviating measures. Mr. Downes made a point that for protection against gamma shine from an unshielded container full of fission products the exclusion area should be approximately three-quarters of a mile. This is for a 500 Mw reactor. He made the point that there is no significant difference in the distance for half versus all of the fission products."
>
> "After considerable discussion the Subcommittee generally agreed that the basic principle of an exclusion area should be for the protection of the public outside of it from the gamma shine. The exclusion distance should be such that for the uniform distribution of 100 per cent (or somewhat less) of the gross fission products within the container the dose at this distance would be [not specified] (according to our notes the Subcommittee did not agree upon any definite number but values of the order of 25 rem and 100 rem were mentioned). Because the greatest part of the dose is delivered in the first hour the actual time to be specified for the accumulation of the dose is not particularly sensitive, but some number should be arrived at. Values for this should range from 4 to 24 hours."

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[10] The Minutes of the ACRS Environmental Subcommittee Meeting, February 18, 1959, Washington, D. C., are included in their entirety as Reference 1, Volume II of this petition and are available in ADAMS Accession No. ML021750385.

[11] 10 CFR PART 20—STANDARDS FOR PROTECTION AGAINST RADIATION

The minutes of the ACRS subcommittee held on August 23, 1960[12], contained a draft of site criteria which defined the basis for an Exclusion Area, an Evacuation Area (later termed the Low Population Zone, and a City Distance (later termed Population center distance) as follows:

> Exclusion Area -- An area whose radius is not less than the distance at which total radiation doses received by an individual fully exposed for two hours to the radioactive consequences of the maximum credible accident would be above 25 R (or equivalent). The area should be under the full control of the applicant. Residents subject to ready evacuation are allowed.
>
> Evacuation Area -- An area whose radius is not less than the distance at which total radiation doses received by an individual fully exposed for the entire maximum credible accident would be above 25 R (or equivalent). Total population not to exceed 10,000 people and no more than 2,000 in any 45° sector.
>
> City Distance -- Distance from reactor to nearest fringe of high density population of a substantial city (above 10,000) which must not be less than distance at which total radiation doses received by a person exposed for the entire maximum credible accident would be above 10 R or equivalent. The real basis, however, for this criterion is an uncontained "puff" release" resulting in a LD-50 dose at the city boundary.

This statement by Dr. Beck that, “The real basis, however, for this criterion is an uncontained puff release of radioactivity resulting in an LD-50 [50 percent chance of death without medical intervention] dose at the city boundary,” relates to the objective stated in the proposed rule that, “Even if a more serious accident (not normally considered credible) should occur, the number of people killed should not be catastrophic.” This statement indicates that the actual criterion in mind was that the distance to the nearest city would be large enough that if the core melted, the containment failed, and all the volatile fission products were released with the wind blowing toward the city, the dose at the city boundary would be that which was estimated to kill half the people exposed to its full effect.[13] The severe accident analysis at the time was WASH 740 which predicted 3,400 acute early fatalities for a worst case reactor accident. [14]

In his testimony at the JCAE Hearings, on Radiation Safety and Regulation, June 12-15, 1961, Mr. Robert Loewenstein, Acting Director, AEC Division of Licensing and Regulations specifically discussed the population center distance as follows[15]:

> "If one could be absolutely certain that no accident greater than the "maximum credible accident" would occur, then the 'exclusion area' and 'low population' zone would provide reasonable protection to the public under all circumstances. There does exist, however,

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[12] Minutes of the ACRS Environmental Subcommittee, August 23, 1960. These minutes are included in their entirety as Reference 3, Volume II of this petition and can be found in ADAMS Accession No. ML021750500.

[13] Nuclear Reactor Safety, On the History of the Regulatory Process, David Okrent, The University of Wisconsin Press, 1981, page 39.

[14] WASH 740, Theoretical Possibilities and Consequences of Major Accidents in Large Nuclear Power Plants, U.S Atomic Energy Commission, March 1957. Maximum of 3,400 lethal exposures for 50 percent release case shown on page 13 of WASH 740.

[15] On the History of the Evolution of Light Water Reactor Safety in the United States, David Okrent, School of Engineering and Applied Science, University of California, ADAMS Accession No.ML090630275.

> a theoretical possibility that substantially larger accidents could occur. It is believed prudent at present, when the practice of nuclear technology does not rest on a solid foundation of extended experience, to provide protection against the most serious consequences of such theoretically possible accidents. Consideration of a 'population center distance' is therefore prescribed: This is a distance by which the reactor would be so removed from the nearest major concentration of people that lethal exposures would not occur in the population center even from an accident in which the containment is breached[16]."

An AEC report on Reactor Site Criteria[17] contained detailed information related to the various considerations involved in the proposed site criteria including the following statements on the basis for the selection of the dose criteria:

> "The end objective in controlling reactor site location is to provide reasonable assurance that the public will not be subjected to undue hazards from operation of the facility. Any meaningful evaluation of the hazard associated with a particular accident must take into account the probability that the accident will occur, the resulting severity of exposures of individual persons to radiation, and numbers of persons at risk. While one cannot make quantitative and detailed evaluation of these factors, the present approach attempts to give to each the greatest consideration presently practicable. The probability of severe accidents is considered to be limited by technical reviews of reactor design and specifications, by conditions of license, and by inspection. Limitations of numbers of persons at risk are provided by exclusion, evacuation, and population center boundaries. Limits imposed on corresponding radiation doses are necessarily arbitrary since the related factors of probability of accident and numbers of persons cannot be closely defined. For the purposes of these criteria we have selected as limits doses which would not result in early manifestations of injury in case of the maximum credible accident and which are believed to involve a reasonably small probability that any individual receiving such a dose would suffer a serious consequence (such as leukemia or cancer) in later years."
>
> "The dose limits specified are 25 rem to the whole body and 300 rem to the adult thyroid. The degree of hazard associated with a dose of 25 rems to the whole body or to a major portion of the body has been qualitatively characterized in a statement by the NCRP that an accidental or emergency dose received only once in the lifetime of a person need not be included in the determination of the exposure status of the person exposed. There is no equivalent recommendation for evaluation of accidental dose to the thyroid. On the basis of staff discussions, 300 r to the adult* thyroid has been used in these criteria."
>
> > **"*** If only adults were involved, the thyroid dose could be much higher. It is currently believed that (1) exposures resulting in a dose of this magnitude to the adult thyroid are likely to result in doses some two or three times as high in very small children; and (2)

[16] A similar statement is included in the AEC Reactor Site Criteria, Report to General Manager by Director, Division of Licensing & Regulations, Document Date May 25, 1959, ADAMS Accession No. ML021960199. The report is included as Reference 4, Volume II of this petition and the statement can be found on page 12 of Reference 4.

[17] AEC Reactor Site Criteria, Report to General Manager by Director, Division of Licensing & Regulations, Document Date May 25, 1959, ADAMS Accession No. ML021960199, included in its entirety as Reference 4 of Volume II of this petition.

> doses of these magnitudes to the thyroid of a small child has some probability of producing cancer of the thyroid in later years.[18]"

In a letter[19] to AEC Chairman John McCone dated October 22, 1960 the ACRS advised against the publication of numerical site criteria as regulations stating that:

> "The Committee believes that the officially endorsed numbers could stifle progress toward a better selection of numbers. The ideas and interpretations from applicants themselves have played a major part in the formulation of the current bases for site evaluation. It would be a significant loss to stop the flow of new ideas from the applicants. The Committee also believes that it is possible that the appearance of quantitative numbers in a Federal regulation or policy statement will reduce the continual awareness of the applicant that he has assumed a responsibility to be alert to and to act on unforeseen disadvantages of a site even after the site has been approved. The Committee, therefore, advises that a quantitative statement of site criteria not be included in Federal regulations."

Regarding the actual numerical values defining criteria for site selection the ACRS included the following statement in a December 13, 1960[20] paper titled, Site Criteria for Nuclear Reactors:

> "The Advisory Committee on Reactor Safeguards believes strongly that there has not yet been a sufficient critical review of the data available to set such numbers as part of a formal regulation. The ACRS recommended a study of the data applicable to the safety problems and the derivation of criteria for all parts of the reactor systems in a letter dated November 16, 1959. As far as the Committee is aware, there has been no such study. Data and numbers applicable to site criteria were suggested as a part of the proposed study. Such a study would permit numbers to be used in defining criteria for site selection. The following numerical values are given as examples to aid in understanding the problem even though their validity is open to question until the study is made."

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[18] A possible source for this reasoning may be from a paper written by Kuper and Cowen: Kuper, JBH and FP Cowan: Exposure Criteria for Estimating the Consequences of a Catastrophe in a Nuclear Plant. Proc. II. Boarding school. Conf. Peaceful Uses of Atomic Energy, vol. 18, p. 319 (1958), Available at https://www.worldcat.org/title/proceedings-of-the-second-united-nations-international-conference-on-the-peaceful-uses-of-atomic-energy-held-in-geneva-1-september-13-september-1958/oclc/9381034. There appears to be an internal inconsistency in this work in that it arbitrarily defines 25 rad of whole body exposure or its equivalent in other types of exposure as the level below which no injury or expense would be expected. As stated previously, at the time the thyroid equivalent to a whole body dose of 25 rad was 150 rad. Notwithstanding this apparent inconsistency the report goes on to suggest extremely high thyroid doses on the order of 2,000 rad as the level below which no immediate injury would be expected in adults, although damage to children or delayed effects in adults is a possibility.

[19] Letter from ACRS to Honorable John A. McCone Chairman U. S. AEC, Subject: Reactor Site Criteria, October 22, 1960. This important letter is included in its entirety on page 21 of Reference 4, Volume II of this petition as Appendix "C-2". This letter was included in the Reactor Site Criteria Report to General Manager by Director, Division of Licensing & Regulations, Document Date May 25, 1959, ADAMS Accession No. ML021960199.

[20] On the History of the Evolution of Light Water Reactor Safety in the United States, David Okrent, School of Engineering and Applied Science, University of California, quoted text appears on page 2-53, ADAMS Accession No. ML090630275.

10 CFR 100.11 and TID-14844

The final rule[21] included the following footnote pertaining to the numerical values used in Part 100 - Reactor Site Criteria:

> 10 CFR Part 100, Footnote 2: A whole body dose of 25 rem referred to above corresponds numerically to the once in a lifetime accidental or emergency dose for radiation workers which, according to NCRP recommendations may be disregarded in the determination of their radiation exposure status (see NBS Handbook 69 dated June 5, 1959). 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 an emergency dose to the public under accident conditions. Rather, this 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 postulated reactor accidents, of exceedingly low probability of occurrence, and low risk of public exposure to radiation[22].

This reference to a once in a lifetime accidental or emergency dose for radiation workers which may be disregarded in the determination of their radiation exposure status as cited in National Bureau of Standards (NBS) Handbook 69 dated June 5, 1959[23] conflicts with 10 CFR Part 20 standards for radiation protection. 10 CFR 20.1201, "Occupational dose limits for adults," states that, "Doses received in excess of the annual limits, including doses received during accidents, emergencies, and planned special exposures, must be subtracted from the limits for planned special exposures that the individual may receive during the current year [annual 20.1201 limits] and during the individual's lifetime [five times 20.1201 limits]."

This reference to NBS Handbook 69 from 1959 resulted in a misinterpretation on the part of radiation protection trainers at several power plants who were instructing that emergency doses could be disregarded in the determination of a worker's radiation status. This confusion was clarified in 1984 with the issuance of information notice (IN) No. 84-40: "Emergency Worker Doses", which clearly states that, "Under current NRC regulations, all occupational doses including emergency doses are required to be included as part of a worker's' exposure history, and hence can affect the workers allowable exposure during the current quarter and subsequent quarters." IN No. 84-40 went on to state that footnote 2 to 10 CFR 100.11(a)(1) had been misinterpreted and that no NRC endorsement of the NBS Handbook 69 emergency dose guidelines/recommendations nor application to 10 CFR Part 20 was ever intended.

The issue of a once in a lifetime dose that could be disregarded in the determination of a worker's radiation exposure status was also addressed in the final rule revising 10 CFR Part 20. It is clearly stated in 56 FR 23372, May 21, 1991, that, "The NRC has not officially sanctioned the 25-rem 'forgivable' emergency dose that has been recommended by some organizations for

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[21]The full text of the Final Rule on Reactor Site Criteria, 27 FRN 3509 1962, April 13, 1962 is included in its entirety as Reference 6, Volume II of this petition.

[22] Appendix D of this petition contains a comparison of the various versions of this footnote in current regulations including 10 CFR 50.34, 10 CFR 50.67, and various subparts of 10 CFR Part 52.

[23]Maximum Permissible Body Burdens and Maximum Permissible Concentrations of Radionuclides in Air and in Water for Occupational Exposure, U. S. Department of Commerce, National Bureau of Standards, Handbook 69, Issued June 5, 1959.

a once-in-a-lifetime dose that would not be counted against the individual's lifetime dose. Consequently, all doses received as [a] result of occupational exposure must be recorded in an individual worker's record."

10 CFR Part 100 makes no similar citation regarding the 300 rem thyroid dose criterion except to state that neither the 25 rem whole body nor the 300 rem thyroid constitute acceptable values for emergency doses to the public under accident conditions. The 300 rem thyroid criterion was not the dose equivalent of 25 rem whole body. Radiation protection standards in effect at the time recommended that occupational iodine exposure for the thyroid be set at a six-to-one ratio to total-body dose.[24] The 25 rem value for emergency exposure for radiation workers was generally accepted by the radiation standards groups but there was no similarly acceptable value for iodine dose to the thyroid, therefore the 300 rem value was somewhat arbitrary. [25]

The final rule included the following note which referenced TID-14844[26] for guidance in developing EAB and LPZ distances stating that:

> "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."

TID-14844 contained a set of assumptions concerning the degree of core damage to be considered in the MCA as well as dose acceptance criteria. TID-14844 fulfilled the need to reduce the subjective manner of arriving at judgment on site suitability which had led to requests to have the AEC make more definitive the basis upon which the data are evaluated and to make more specific the safety criteria which govern the AEC's consideration of site suitability[27].

The staff analysis of the final rule concluded that it was "intended to be an interim measure until the state of the art allows for more definitive standards to be developed."[28]

The Final Rule included the following statement in Paragraph 100.1;

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[24] The maximum permissible average concentrations of radionuclides in air and water are determined from biological data whenever such data are available, or are calculated on the basis of an averaged annual dose of 15 rems for most individual organs of the body, 30 rems when the critical organ is the thyroid or skin, and 5 rems when the gonads or the whole body is the critical organ. NBS Handbook 69, 1959.

[25] George T. Mazuzan & J. Samuel Walker, "Controlling the Atom: The Beginnings of Nuclear Regulation, 1946-1962, November 15, 1984, page 238.

[26] Technical Information Document (TID)-14844, "Calculation of Distance Factors for Power and Test Reactor Sites," USAEC, March 23, 1962. ADAMS Accession No. ML021720780.

[27] Paraphrased from Summary Item 4 of the Atomic Energy Commission, Reactor Site Criteria, Report to the General Manager by the Director, Division of Licensing and Regulation, Document Date May 25, 1959, ADAMS Accession No. ML021960199. The statement concerning the need for more definitive criteria can be found on page 1, paragraph 4, of Reference 4, Volume II of this petition.

[28] George T. Mazuzan & J. Samuel Walker, "Controlling the Atom: The Beginnings of Nuclear Regulation, 1946-1962, November 15, 1984, page 245.

> (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.

Conclusions Regarding 10 CFR Part 100 and TID-14844

The site criteria formalized in 10 CFR Part 100 were designed so that they would not conflict with the siting decisions that had already been made by the AEC. David Okrent included the following statement in his book, "Nuclear Reactor Safety, On the History of the Regulatory Process:"

> "Reminiscing almost 20 years later, regulatory staff members, who had worked on this draft, recalled trying to find a set of parameters and assumptions which would fit essentially all the previously approved reactor site combinations, within some broader, generally acceptable framework.[29]"

The purpose of TID-14844 is clearly stated on page 1 of the document as follows:

> "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) can be used to calculate distance requirements for reactor sites which are generally consistent with current siting practices."

The 25 rem whole body dose criterion was based on concerns for the external gamma shine from fission products contained in the reactor building. TID-14844, Assumption 11 states the following:

> "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."

The 25 rem whole body criterion was not the limiting concern for any large reactor or for any reactor with a shielded containment. TID-14844 identified two radiological concerns: (1) the direct shine from unshielded containment structures which were common at the time; and, (2) the thyroid dose from the inhalation of iodine. Examination of TID-14844, Figure 1, Exclusion Radius Determination (included on page 4 of Appendix A, Volume I of this petition), reveals that the 25 rem whole body limitation controlled the siting determination for unshielded reactors with a thermal power level less than 300 megawatt thermal (MWt). For all reactors with a power

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[29] Nuclear Reactor Safety, On the History of the Regulatory Process, David Okrent, The University of Wisconsin Press, 1981, footnote, page 39.

level above 300 MWt, the 300 rem thyroid criterion controlled the siting determination.[30] Based on the petitioner's experience in performing dose consequence analyses, modern power reactors were governed by the limitation of the 300 rem thyroid dose criterion and not the 25 rem whole body dose criterion[31]. This fact is especially important in relation to the development of the TEDE dose criteria which will be discussed in more detail in later sections of this petition.

Control Room Dose Criterion: Objectives

In 1971 Appendix A, "General Design Criteria for Nuclear Power Plants," was added to 10 CFR Part 50. General Design Criterion 19 (GDC-19) specified that adequate protection shall be provided to permit access and occupancy of the control room for the duration of an accident without exceeding a radiation exposure of 5 rem whole body or its equivalent to any part of the body. From its inception, GDC-19 became the limiting dose criteria in almost all radiological dose consequence analyses. The control room accident dose criterion corresponds numerically to the current yearly occupational dose limit for routine operations. At the time GDC-19 was established, the maximum occupational dose limit was 3 rem per quarter with a lifetime limit of 5(N-18) where N was the workers age in years. The regulations in effect at that time allowed for a maximum yearly total of 12 rem providing that the occupational life time average did not exceed 5 rem per year[32]. Currently the occupational yearly dose limit for routine operations is 5 rem total effective dose equivalent (TEDE) with an organ dose limit of 50 rem.

There are no footnotes or notes in criterion 19 to define the accident condition to be analyzed as is the case in 10 CFR 100.11[33]. By guidance, licensees are directed to analyze the control room radiological habitability with the same conservative assumptions and MCA source term used in the evaluation of the off-site reference values.

The general design criteria in Appendix A were developed and issued to establish minimum necessary design, fabrication, construction, testing, and performance requirements for structures, systems, and components that provide reasonable assurance that the facility can be operated without undue risk to the health and safety of the public. The criteria were published for public comment as a proposed amendment to Part 50 (32 FR 10213; July 11, 1967). Early versions of the control room criteria specified that adequate radiation protection shall be provided to permit access, even under accident conditions, to equipment in the control room or other areas as necessary to shut down and maintain safe control of the facility without radiation exposures of personnel in excess of 10 CFR 20 limits. Following resolution of public comments, the criteria were published as a final rulemaking (36 FR 3255; February 20, 1971). In different

[30] Appendix A, Volume I of this petition contains a more detailed discussion of the influence of the siting of early reactors with unshielded containment structures on design basis accident dose criteria.

[31] This statement is based on the petitioner's 40 plus years of performing and reviewing dose consequence analyses.

[32] Reference 11, Volume II of this petition entitled, Excerpts from, A Brief History of Radiation - Protection Standards, William C. Inkret, Charles B. Meinhold, and John C. Taschner, *Los Alamos Science,* Number 23 1995. Available at https://permalink.lanl.gov › object › lareport › LA-UR-95-4005-04, demonstrates how recommendations for safe levels of radiation exposure have reduced substantially as scientists learned more about the health effects of radiation.

[33] 10 CFR 100.11, Footnote 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.

versions of the criteria, the control room criteria were designated as GDC-11, 14, 15, and 17 and, in the final rulemaking, GDC-19. The current GDC-19 provides:

> A control room shall be provided from which actions can be taken to operate the nuclear power unit safely under normal conditions and to maintain it in a safe condition under accident conditions, including loss-of-coolant accidents. Adequate radiation protection shall be provided to permit access and occupancy of the control room under accident conditions without personnel receiving radiation exposures in excess of 5 rem whole body, or its equivalent to any part of the body, for the duration of the accident. Equipment at appropriate locations outside the control room shall be provided (1) with a design capability for prompt hot shutdown of the reactor, and controls to maintain the unit in a safe condition during hot shutdown, and (2) with a potential capability for subsequent cold shutdown of the reactor with suitable procedures.
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> Applicants for and holders of construction permits who apply on or after January 10, 1997, applicants for design certifications under part 52 of this chapter who apply on or after January 10, 1997, applicants for and holders of combined licenses under part 52 of this chapter who do not reference a standard design certification, or holders of operating licenses using an alternative source term under § 50.67, shall meet the requirements of this criterion, except that with regard to control room access and occupancy, adequate radiation protection shall be provided to ensure that radiation exposures shall not exceed 0.05 Sv (5 rem) total effective dose equivalent (TEDE) as defined in § 50.2 for the duration of the accident.

The explicit basis for the selection of 5-rem whole body or its equivalent criteria is not described in the statements of consideration (SOC) for the 1971 rule, which published the GDCs. The SOCs addressed the criteria in the aggregate; the individual criteria were not discussed. Notwithstanding the lack of the documentation of the reasoning behind the selection of the criteria, it is generally understood that the objective of the criteria was to ensure that the design of the control room and its habitability systems would provide a “shirt-sleeved” environment for the control room operators. Such an environment was perceived to be supportive of facilitating operator response to normal and accident conditions and would minimize errors of omission or commission.

Since a whole body radiation exposure of 5 rem was comparable to the occupational dose limits, it can be proffered that this value was considered unlikely to cause increased anxiety potentially resulting in operator impairment. At the time that GDC-19 was being published, Part 20 limited occupational radiation exposure to 3 rem whole body dose per calendar quarter provided that the individual’s cumulative dose history did not exceed 5 rem per year for each year of exposure after the age of 18. Therefore, it was possible to receive a radiation exposure of up to 12 rem in a given year provided that the individual’s cumulative dose history did not exceed 5 rems times the individual’s age in year’s age (N) minus 18 or 5 x (N-18).

At the time that GDC-19 was being published the philosophy of maintaining radiation exposures as low as is reasonably achievable (ALARA) was not treated as a practice required by regulation. As shown by an examination of Table 4.4 from NUREG-0713,[34] although rare today, occupational exposures of 5 rem and higher were not unusual in the 1970’s. It should be noted

[34] Occupational Radiation Exposure at Commercial Nuclear Power Reactors and other Facilities, Available at https://www.nrc.gov › doc-collections › nuregs › staff › sr0713.

that GDC-19 is a design criterion and does not displace the radiation protection standards of Part 20. The radiation exposure of control room operators is treated, as for any radiation worker at the facility, as occupational exposure under Part 20.[35]

On May 17, 2007 Raymond Crandall, a long term nuclear professional specializing in radiological analyses, submitted a petition for rule change to eliminate the performance-based control room dose criteria[36]. The petitioner proposed as an alternative to a dose-based acceptance criterion, the following guidelines based on "good engineering principles."

> "As an example, the guidance could include requirements such as:
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> - The control room ventilation system should isolate on the detection of high radiation or toxic gas intake.
> - The control room should have a minimum of one foot of concrete shielding (or equivalent) on all surfaces.
> - Self Contained Breathing Apparatus (SCBA' s) and potassium iodide (KI) tablets should be readily available for operator use. Operators should maintain training in SCBA's.
> - Procedural controls to maintain a low leakage boundary, such as preventative maintenance/routine inspection of door seals and dampers should be implemented.
> - Procedures should be developed to ensure control room purging is considered when the outside concentration is less than the inside concentration.
> - Existing emergency filtration systems should be maintained to practical performance criteria"

The petitioner did not proffer an alternative dose acceptance value but did include the following discussion of the NRC's DBA dose acceptance criteria:

> "Third, the dose limit itself is overly restrictive. Why should the public be allowed to receive 25 REM TEDE and the control room operator be limited to 5 REM? There is no health consequence to a dose of 25 REM, and the EPA protective action guidelines would allow such a dose for control room operator functions. In the past, in an attempt to find some safety significance to the control room habitability requirements, the NRC staff has stated that the operators may not feel adequately protected to perform their function if the plant conditions and design analyses did not demonstrate that the 5 REM limit could be met. The control room operator is a trained nuclear professional,

[35] Although the scope of Part 20 does not specifically address radiation protection standards during emergency conditions, it doesn't specifically exclude emergency conditions either. Information Notice No. 84-40, Emergency Worker Doses, reminded licensees of their obligation to include doses received during emergency conditions in determining compliance with the occupational dose limits.

[36] PRM-50-87, To Eliminate CR Accident Dose Criteria, is included in its entirety as Reference 8, Volume II of this petition and is available in ADAMS Accession No. ML071490250.

dedicated to the protection of public safety, and would be willing to receive a dose higher than 5 REM to mitigate an accident."

On January 26, 2009 the NRC published, "Raymond A. Crandall; Denial of Petition for Rulemaking[37]. The basis for the denial included the NRC's preference for performance-based regulations that do not specify the exact methods which licensees must follow in order to meet a particular regulation. The denial did not specifically address the comparison between the control room and the public acceptance criteria. Both the petition and the denial are included as Reference 8 and Reference 9 of Volume II of this petition and provide additional information regarding GDC-19.

Additional Challenges to Meeting the Requirements of GDC-19

As can be seen by examination of representative MCA results shown in Appendix E[38] of this petition, many licensees' evaluations have a relatively small margin to the control room acceptance value. With the adoption of the TEDE dose criterion many licensees have gained operational flexibility over the previous use of a thyroid dose criterion. The current thyroid dose weighting factor being used in the calculation of TEDE is 0.03 per 10 CFR 20.1003. The International Commission of Radiation Protection (ICRP) Publication 103 has recommended the use of a thyroid weighting factor of 0.04. The NRC's Office of Nuclear Regulatory Research completed a study entitled, "Control Room Dose Evaluation Using ICRP 103 Dose Conversion Factors," letter report (ADAMS Accession No. ML17156A603), which concludes that: "Application of the ICRP 103 DCFs will result in an increase in the range of 23 to 25% in the TEDE doses for the control room." The degree of impact will depend on the amount of credit taken for various iodine removal mechanisms both natural and engineered. However, if the ICRP recommendations are ever incorporated into NRC's regulations and guidance, the incorporation of a thyroid weighting factor of 0.04 will decrease the already small margin many licensees have in their control room dose consequence analysis.

GDC-19 requires that, "*Adequate radiation protection shall be provided to permit access and occupancy of the control room under accident conditions without personnel receiving radiation exposures in excess of 5 rem whole body, or its equivalent to any part of the body, for the duration of the accident."* The NRC has not emphasized the issue of control room access in any of the regulatory guides dealing with control room habitability. As such most licensees do not include an evaluation of access dose in their control room dose consequence analysis. Including access dose in the calculation of the total control room would decrease the already small margin most licensees have in their control room dose consequence analysis.

Conclusions Regarding GDC-19

The objective of GDC-19 appears to have been based on the concept of providing a comfortable environment for the operators. The acceptance criterion of 5 rem is well below the threshold value that could have a significant impact on the health and safety of a control room operator. The question for consideration in this petition is whether a slightly higher value could still satisfy the objective of providing a comfortable environment for the operator while reducing the regulatory burden associated with demonstrating compliance with the current criteria.

[37] (NRC-2007-0016; PRM-50-87), Raymond A. Crandall; Denial of Petition for Rulemaking, is included in its entirety as Reference 9, Volume II of this petition.

[38] Appendix E of this petition is a compilation of the present calculated MCA dose evaluations as described in licensee's final safety analysis reports.

NUREG-0625 and 10 CFR 50.34

In August 1978, the Nuclear Regulatory Commission directed the staff to develop a general policy statement on nuclear power reactor siting which resulted in NUREG-0625[39], "Report of the Siting Policy Task Force." NUREG-0625 recommended that fixed distances should be required for the EAB and the LPZ.

ABSTRACT [From NUREG-0625]

In August 1978, the Nuclear Regulatory Commission directed the staff to develop a general policy statement on nuclear power reactor siting. A Task Force was formed for that purpose and has prepared a statement of current NRC policy and practice and has recommended a number of changes to current policy. The recommendations were made to accomplish the following goals:

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

2. To take into consideration in siting the risk associated with accidents beyond the design basis (Class 9) by establishing population density and distribution criteria. Plant design improvements have reduced the probability and consequences of design basis accidents but there remains the residual risk from accidents not considered in the design basis. Although this risk cannot be completely reduced to zero, it can be significantly reduced by selective siting.

3. To require that sites selected will minimize the risk from energy generation. The selected sites should be among the best available in the-region where new generating capacity is needed. Siting requirements should be stringent enough to limit the residual risk of reactor operation but not so stringent as to eliminate the nuclear option from large regions of the country. This is because energy generation from any source has its associated risk, with risks from some energy sources being greater than that of the nuclear option.

The concern was that siting practices were not providing enough emphasis on site isolation as an important contributor to defense in depth because ESF systems such as iodine filters, containment sprays, and double containment structures could be designed to make almost any site acceptable from an accident dose calculation point of view.

As shown below described in the Background to the Final Rule on, "Reactor Site Criteria Including Seismic and Earthquake Engineering Criteria for Nuclear Power Plants," Federal Register Volume 61, Number 239, Wednesday, December 11, 1996, in the 1980 Authorization Act for the NRC, the Congress directed the NRC to decouple siting from design and to specify demographic criteria for siting.

[39] NUREG-0625 Report of the Siting-Policy Task Force, Office of Nuclear Reactor Regulation U.S. NRC, ADAMS Ascension No. ML12187A284.

Background

The present regulation regarding reactor site criteria (10 CFR Part 100) was promulgated April 12, 1962 (27 FR 3509). NRC staff guidance on exclusion area and low population zone sizes as well as population density was issued in Regulatory Guide 4.7, “General Site Suitability Criteria for Nuclear Power Stations," published for comment in September 1974. Revision 1 to this guide was issued in November 1975. On June 1, 1976, the Public Interest Research Group (PIRG) filed a petition for rulemaking (PRM-100-2) requesting that the NRC incorporate minimum exclusion area and low population zone distances and population density limits into the regulations. On April 28, 1977, Free Environment, Inc. et al., filed a petition for rulemaking (PRM-50-20). The remaining issue of this petition requests that the central Iowa nuclear project and other reactors be sited at least 40 miles from major population centers. In August 1978, the Commission directed the NRC staff to develop a general policy statement on nuclear power reactor siting. The “Report of the Siting Policy Task Force" (NUREG-0625) was issued in August 1979 and provided recommendations regarding siting of future nuclear power reactors. In the 1980 Authorization Act for the NRC, the Congress directed the NRC to decouple siting from design and to specify demographic criteria for siting. On July 29, 1980 (45 FR 50350), the NRC issued an Advance Notice of Proposed Rulemaking (ANPRM) regarding revision of the reactor site criteria, which discussed the recommendations of the Siting Policy Task Force and sought public comments. The proposed rulemaking was deferred by the Commission in December 1981 to await development of a Safety Goal and improved research on accident source terms. On August 4, 1986 (51 FR 23044), the NRC issued its Policy Statement on Safety Goals that stated quantitative health objectives with regard to both prompt and latent cancer fatality risks. On December 14, 1988 (53 FR 50232), the NRC denied PRM-100-2 on the basis that it would unnecessarily restrict NRC's regulatory siting policies and would not result in a substantial increase in the overall protection of the public health and safety. The Commission is addressing the remaining issue in PRM-50-20 as part of this rulemaking action.

This was accomplished by relocating source term and dose calculations to 10 CFR 50.34 (61 FR 65157 December 11, 1996). The proposed rule (57 FR 47802) decoupled siting from accident source term and dose criteria by applying fixed distances for the EAB and LPZ based on Regulatory Guide 4.7, “General Site Suitability for Nuclear Power Stations”. After numerous comments stating that the source term and dose calculations should be retained, the commission decided to reinstate dose criteria in the second proposed rule (59 FR 52255) to establish the requirements for the EAB and the outer boundary of the LPZ. The reference to TID-14844 was removed and the revised dose standard was expressed in TEDE. The acceptance criterion for thyroid dose was eliminated from the regulation. The Part 100 values were used as a starting point for the revised dose criteria. The petitioner has found no evidence that the Part 100 criteria were ever re-examined as being appropriate for continued use or for their conversion to a single criterion based solely on TEDE. The only consideration documented in the statements of consideration was how to translate the 25 rem whole body and 300 rem thyroid dose criteria to a single TEDE dose reference value. The nuclear industry suggested a value of 34 rem based on a thyroid weighting factor of 0.03.[40] The NRC staff

[40] The nuclear industry suggested that the TEDE equivalent for the Part 100 limits should be 34 rem. This is based on adding the 25 rem whole body dose limit to the thyroid dose limit adjusted using a thyroid weighting factor of 0.03; 25 + (300 x 0.03) = 34.

suggested a value of 27[41] rem based on a consideration of the resulting latent cancer fatality risk. This value was rounded down to 25 rem TEDE and adopted in the rule. The use of risk terminology in the statements of consideration may have led some to believe that the 25 rem TEDE value was based on the risk of latent cancer fatality however this is not the case. The § 100.11 values were based on considerations of a non-stochastic dose not on stochastic latent cancer fatality. In the opinion of the petitioner the conversion of deterministic values to latent cancer fatalities, adding the results, then converting back implying that the resulting criterion is based on latent cancer risk is not good science. The final rule (61 FRN 65157) did acknowledge that the 300 rem thyroid was the limiting dose criterion in licensing reviews and that the 25 TEDE criterion represented a relaxation of the dose criterion.[42] However, the full extent of the relaxation may not have been adequately addressed in the statement of considerations accompanying the final rule.

These discussions did not recognize the fact that the 25 rem whole body dose criterion was specifically based on concerns for direct gamma shine from unshielded containments and that this criterion had no influence on the siting of any modern power reactor. For all practicable purposes there was only one criterion (the 300 rem thyroid dose from inhalation of iodine) that had any influence on the siting of modern power reactors. If the intent was to maintain the siting practices of the existing reactors, then either an organ dose limitation should have been included in the regulation or a TEDE value should have been chosen such that the associated thyroid dose would be limited to approximately 300 rem. Instead the 25 rem value, which never impacted the siting of any modern power reactor under 10 CFR 100.11, was now the sole TEDE criterion for all new reactors and for licensees adopting the AST.

TEDE dose criterion and the elimination of the thyroid dose value

The following excerpt is from SECY-98-154[43], Results of the Revised (NUREG-1465) Source Term Rebaselining for Operating Reactors, Attachment, “Fission Product Source Term Rebaselining,” (page 25 of the attachment). The purpose of this work was, “To provide the Commission with the results and findings of an evaluation of the impact of implementing the revised source term described in NUREG-1465, “Accident Source Terms for Light-Water Nuclear Power Plants,” for operating reactors.” This was done to explore the differences that would result in performing accident dose calculations with the revised source term. The work showed that the use of the revised source term generally resulted in the calculation of lower doses.

> Increased Containment Leak Rate
>
> For this potential plant change, rebaselining calculations were done for all three rebaselining plants. For each plant the containment leak rate was arbitrarily increased

[41] 59 FR 52255 contains a discussion of the derivation of the 27 rem equivalency. The 25 rem WB was equated to a latent cancer fatality risk of .025 based on the risk coefficient from BEIR doubled for short term exposure (5.0E-4 x 2 = 0.001 per rem); 300 rem thyroid was equated to a latent cancer fatality of 0.002. The Thyroid risk coefficient appears to be from NUREG/CR-2414 (6.4 per 1,000,000 x 300 = 0.0019 rounded to 0.002). The individual risks were added for a total cancer risk of 0.027. Then a risk coefficient of 0.001 per rem was used convert the risk back to a dose of 27 rem. This value was then rounded to the final 25 rem TEDE.

[42] Relevant excerpts from 59 FRN 52255 and 61 FRN 65157 are included as Reference 7, Volume II of this petition.

[43] SECY-98-154, Results of the Revised (Nureg-1465) Source Term Rebaselining for Operating Reactors, June 30, 1998, ADAMS Ascension No. ML992880064.

by a factor of 10. The results of these calculations for Zion are given in Table 26 along with the Phase III results. For Zion, the containment leak rate was increased from 0.1%/day to 1%/day, and the doses increased by a factor of 10. If the EAB TEDE dose is limiting for Zion, a factor of three increase in the containment leak rate would be the highest allowable to meet the proposed TEDE dose limit. Doses for Surry and Grand Gulf increased by a factor of 10, except for Grand Gulf thyroid doses which only increase by a factor of eight due to MSIV leakage.

SECY-98-154 Table 26. Increased Containment Leak Rate for Zion

Case	EAB			LPZ		
	Thyroid	Whole Body	TEDE	Thyroid	Whole Body	TEDE
Phase III[44]	145(.3h)	1.94(.9h)	**8.02**(.5h)	47.6	1.04	3.06
Increased Leak Rate	1398(0h)	16.2(.5h)	74.2(0h)	448	10.4	29.4

The text states that, “If the EAB TEDE dose is limiting for Zion, a factor of three increase in the containment leak rate would be the highest allowable to meet the proposed TEDE dose limit.” If the Phase III TEDE dose was adjusted by a factor of 3.12, (25 ÷ 8.02), to reach the EAB TEDE design criterion of 25 rem, the corresponding thyroid dose would be approximately 450 rem.

Therefore, because of the elimination of the 10 CFR 100.11 thyroid dose value of 300 rem, there is a considerable reduction in the conservatism associated with meeting DBA dose acceptance criterion. The increased margin based solely on the conversion to a TEDE dose criterion above and beyond that associated with other aspects of the AST such as the timing of the release into containment, can be substantial depending on the individual case under consideration. The conversion of the dual dose criteria of 10 CFR 100.11 into a single TEDE dose criterion allows for thyroid doses well in excess of the previously controlling 300 rem criterion.

As previously stated, the promulgation of Part 100 and its technical basis document TID-14844 was based on the concept that the dose profiles of future reactors should roughly reflect the dose profiles of the plants that were already operating at the time of its inception. Part 100 made reference 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 thyroid dose was limiting for all large reactors and minimized the significance of the 25 rem whole body dose restriction. New reactors and existing plants that adopt the alternative source term can demonstrate compliance with the 25 rem TEDE dose criterion while substantially exceeding the previously limiting 300 rem thyroid dose and thereby are allowed by regulation to exceed the MCA dose profiles of the earliest reference reactors.

[44] SECY-98-154 explored many different cases to evaluate the revised source term with existing Standard Review Plan (SRP) treatment of the removal mechanisms as well as updated removal models. The Zion FSAR models the containment as a single sprayed volume. The Zion FSAR assumes a spray removal rate significantly larger than the Standard Review Plan. In Phase III, removal rates from the SRP and from the updated models in NUREG/CR-5966 were used.

10 CFR 50.67, "Accident source term"

10 CFR 50.67 (64 FR 71990 December 23, 1999) restated the 50.34 off-site dose criteria and included a revised control room design criterion of 5 rem TEDE, again eliminating the organ dose value. As in 10 CFR 50.34, the rule does not reference TID-14844 or any specific source term but retained the general reference to a source term resulting from a substantial meltdown of the core with a subsequent release of appreciable quantities of fission products. Employing 10 CFR 50.67, licensees can voluntarily revise their dose consequence analyses using guidance from regulatory guide (RG) 1.183 "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Plants." Many licensees have submitted license amendment requests to adopt the alternative source term (AST) because it allows greater flexibility in meeting accident dose acceptance criteria. The use of the alternative source term provides some additional margin for licensees due to the timing of the release into containment and the chemical form of the iodine released into containment. A significant additional margin is attained though the adoption of the AST because the dose criteria specified in 10CFR 50.67 are expressed in TEDE only with no additional organ specific dose value. Part of the reason that many licensees are adopting 10 CFR 50.67 is due to the elimination of the thyroid dose component which was almost always the limiting consideration in accident dose consequence analyses. Under current NRC regulations an acceptable radiological design, as calculated under accident conditions for a new reactor, could result in EAB thyroid doses exceeding 500 rem in 2 hours while still meeting the reference value of 25 rem TEDE.

Part 100 vs. GDC-19

The comparison between the accident control room design criterion of 5 rem and the accident off-site reference value for the general population of 25 rem contradicts one of the underlining principals of 10 CFR Part 20 standards of radiation protection which have always limited the dose to the general population to at least one tenth that for the radiation worker. Part 20 standards make a clear distinction between adult occupational dose limits and dose limits for minor occupational workers and members of the public. 10 CFR 20.1201 limits the annual occupational dose to individual adults to 5 rem TEDE in a year. 10 CFR 20.1207 limits the annual occupational dose to minors (individuals less than 18 years old) to 10% of the annual limits specified for adult workers. 10 CFR 20.1301 limits the dose to individual members of the public from the licensed operation of a nuclear power reactor to 0.1 rem TEDE in a year.

The significant difference in the adult occupational dose limit as compared to the limit for the public is appropriate and reasonable based on many factors. All nuclear power radiation workers undergo extensive and continuous training and are required to be knowledgeable concerning the biological effects of ionizing radiation as well as in the use of various protective measures available to limit dose. Nuclear workers have chosen to accept the risk associated with incurring occupational radiological exposure in association with their daily work. In addition, the nuclear worker has ready access to well-trained experts in radiation protection by the health physics staff personnel that are always present on site. In addition, control room personnel are trained and equipped with personnel protective equipment should their use be deemed necessary during an accident condition.

In contrast, members of the public generally have little or no training concerning the biological effects of ionizing radiation or in the basic actions which could be taken to limit exposure. Perhaps more significant is the fact that members of the public include pregnant woman, infants and children whose increased radio-sensitivity compared to adults is well documented.

It should be noted that other regulatory accident dose criteria such as § 72.106 Controlled area of an ISFSI [Independent Spent Fuel Storage Installation] or MRS [Monitored Retrievable Storage] and § 70.61 Performance requirements (for certain licensees authorized to possess a critical mass of special nuclear material), limit the dose to individuals outside the controlled area to less than (§ 70.61) or equal to (§ 72.106) the accident dose criterion for facility workers. The accident dose acceptance criteria for power reactors is the only instance in regulation where the criteria for occupational workers are lower than the criteria for the public.

In addition to the fact that power reactor design basis accident acceptance criteria for the public are a factor of 5 higher than the acceptance criterion for the dose to the occupational worker in the control room, the 25 rem value for the exclusion area boundary (EAB) is for a two hour period as opposed to the 5 rem control room value which is calculated for the duration of the accident (defined in guidance as a 30 day period). This difference in the duration associated with the respective criteria further aggravates the discrepancy between the design basis dose criteria in the current regulations. Basic health physics teaches us that for a given absorbed dose, the biological damage caused by ionizing radiation is significantly enhanced if the dose is incurred over a shorter time.

The guidance offered by the Food and Drug Administration (FDA) regarding the recommendations for issuance of potassium iodide (KI)[45] to the public during a radiological emergency demonstrates the radio-sensitivity of children for exposure to radioactive iodine. For adults over the age of 40 the FDA recommends the administration of KI at a thyroid threshold exposure of 500 rem; for ages 19 through 40 the thyroid exposure threshold is 10 rem, and for pregnant or lactating women and children the thyroid threshold exposure is 5 rem.

The argument that is most often postured to explain the difference in the design accident dose criteria for the control room and the general population is that the control room operators cannot evacuate whereas members of the public can. An effective evacuation of the LPZ taking place within two hours, although not stated, appears to be an implicit assumption imbedded in the regulation which can serve to limit the immediate non-stochastic health effects in the LPZ during the postulated MCA. There is no long-term accident dose design criterion within the LPZ. There is only a short term 2-hour value at the EAB and a long-term value at the outer boundary of the LPZ. There is no 4, 6 or 8-hour design criterion within the LPZ. If for instance a licensee calculates a dose of 25 rem at the EAB for the required 2-hour time period, the integrated dose within the LPZ could be significantly higher than 25 rem for longer time periods provided that the integrated dose at the outer boundary of the LPZ does not exceed 25 rem for the duration of the accident.

The Environmental Protection Agency (EPA) protective action guides (PAGs) for evacuation are often cited as the basis the allowable design criteria of 25 rem for members of the public. The EPA PAGs[46] recommend sheltering-in-place or evacuation for a projected TEDE dose from 1 to 5 rem aggregated over 4 days and recommend the administration of KI for a projected child thyroid dose of 5 rem from radioactive iodine. These more reasonable dose guidelines for members of the public are sometimes used to justify the much higher design basis accident dose criteria in current NRC regulations.

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[45] Guidance Potassium Iodide as a Thyroid Blocking Agent in Radiation Emergencies, U.S. Department of Health and Human Services, Food and Drug Administration, Center for Drug Evaluation and Research (CDER), December 2001, page 6.

[46] PAG Manual Protective Action Guides and Planning Guidance For Radiological Incidents, U.S. Environmental Protection Agency, Final Revision, Table 1-1, EPA-400/R-17/001, January 2017.

However, what is missing from these arguments is that both the off-site and the control room design basis accident dose criteria should be viewed as figures of merit that are used to test the ruggedness of engineered safety feature systems. As such, these values have a direct impact on the design of a nuclear power plant and are essential for maintaining effective licensing bases and technical specifications (TSs) on a variety of reactor operation parameters such as safety limits, containment integrity and the performance of engineered safety systems. These values need to have a clear and defendable basis which should emphasize the protection of the public above all other considerations. Currently, by NRC regulation the control room operator has a universally acceptable accident dose design criterion that is the same as the current annual occupational dose limit for routine operations. By contrast the accident dose design criterion for an individual at the EAB is five times higher in magnitude and can be incurred at a normalized dose rate that is 1,800 times higher than for the control room operator.[47] Further, as explicitly stated in the regulation,[48] "neither its use [25 rem] 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 an emergency dose to the public under accident conditions."

Based on a review of the history of the current accident dose criteria, it appears that the public criteria were based on the avoidance of somatic damage, that is the avoidance of readily observable demonstrable changes in organs or cells. In contrast, the significantly lower control room criterion appears to have been derived from consideration of stochastic effects, that is the consideration of random events, the probability of which increases with an increase in radiation dose. In the opinion of the petitioner, this discrepancy is difficult to justify.

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[47] The control room dose limit of 5 rem is for the duration of the accident which by guidance has been defined as 30 days or 720 hours. In contrast the 25 rem limit at the EAB is for a 2 hour period. For simplification if we assume a constant dose rate over the respective time periods the allowable dose rate for an individual at the EAB is 1,800 times the dose rate for the control room operator. (25 rem/2 hours = 12.5 rem/hour; 5 rem/720 hours = 0.006944 rem/hour)

[48] 10 CFR 100.11, Footnote 2: A whole body dose of 25 rem referred to above corresponds numerically to the once in a lifetime accidental or emergency dose for radiation workers which, according to NCRP recommendations may be disregarded in the determination of their radiation exposure status (see NBS Handbook 69 dated June 5, 1959). 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 an emergency dose to the public under accident conditions. Rather, this 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 postulated reactor accidents, of exceedingly low probability of occurrence, and low risk of public exposure to radiation.

Current Health Physics Guidance

The Environmental Protection Agency (EPA) recommends 25 rem as an appropriate emergency worker guideline for saving a life or the protection of large populations and explains that:

> "The 25 rem (250mSv) lifesaving response worker guidelines provide assurance that exposures will not result in detrimental deterministic health effects (i.e., prompt or acute effects). However, it could increase the risk of stochastic (chronic) effects, such as the risk of cancer." [49]

The international commission on radiological protection's (ICRP) 2007 recommendations[50] states the following concerning the factors influencing the choice of source-related dose constraints and reference levels:

> "At doses higher than 100 mSv [10 rem], there is an increased likelihood of deterministic effects and a significant risk of cancer. For these reasons, the Commission [ICRP] considers that the maximum value for a reference level is 100 mSv incurred either acutely or in a year. Exposures above 100 mSv incurred either acutely or in a year would be justified only under extreme circumstances, either because the exposure is unavoidable or in exceptional situations such as the saving of life or the prevention of a serious disaster. No other individual or societal benefit would compensate for such high exposures."

The following excerpts are taken from ICRP Publication 109, "Application of the Commission's Recommendations for the Protection of People in Emergency Exposure Situations."

> "The Commission has recommended that reference levels for emergency exposure situations should typically be set in the band of 20–100 mSv (acute or per year)."

> "While reference levels for emergency exposure situations may be fixed at values up to 100 mSv (ICRP, 2007, Table 5), they would only be set at the upper end of the band 20–100 mSv in unusual or extreme circumstances where actions taken to reduce exposures would be very disruptive."

The following excerpt is from section 3.71 of IAEA Safety Standards for protecting people and the environment, "Radiation Protection of the Public and the Environment," General Safety Guide No. GSG-8, International Atomic Energy Agency, Vienna, 2018.

> "For emergency exposure situations, GSR Part 3 [2] and GSR Part 7 [4] require that a reference level expressed in terms of residual dose be set, typically as an effective dose in the range of 20–100 mSv, acute or annual, that includes dose contributions via all exposure pathways. The residual dose is the dose expected to be incurred after protective actions have been terminated (or after a decision has been taken not to take protective actions) and so is the dose accumulated from the initiation of the event, through a specified period of time. The purpose of a reference level in an emergency exposure situation is to guide the optimization process of protection strategies aimed at reducing the doses to be incurred by individuals and to be a benchmark for a retrospective assessment of the

[49] PAG Manual Protective Action Guides and Planning Guidance For Radiological Incidents, U.S. Environmental Protection Agency, Final Revision, Table 1-1, EPA-400/R-17/001, January 2017.

[50] ICRP Publication 103, Section 5.9.3, "Factors influencing the choice of source-related constraints and reference levels", paragraph 236.

> effectiveness of protective actions taken and the protection strategy in an emergency response."

The following statement is taken from the NRC web page, "Radiation Exposure and Cancer"

> "Although radiation may cause cancer at high doses and high dose rates, public health data do not absolutely establish the occurrence of cancer following exposure to low doses and dose rates — below about 10,000 mrem (100 mSv)."

The following statement is taken from, "A Perspective on Risk to the Fetus from Ionizing Radiation," Duke University and Duke Medicine, Radiation Safety Division, www.safety.edu.

> "Fetal Dose Exceeding 10,000 millirem -- The lower limits (in terms of statistical confidence intervals) for threshold doses for effects such as mental retardation and diminished IQ and school performance fall within this range. Overall, exposure at levels exceeding 10 rem could be expected to result in a dose-related increased risk for deleterious effects. For example, the lower limit (95% confidence interval) for the threshold for mental retardation is about 15 rem, which an expectation value of about 30 rem."

The following excerpt is from the Health Physics Society Radiation Fact Sheet on Radiation Exposure and Pregnancy, June 2017:

> Potential radiation effects vary depending on the stage of fetal development and on the magnitude of the radiation doses received. According to NCRP Report No. 174 (NCRP 2013), doses below 100 mSv should not increase the risk of reproductive effects (birth defects or miscarriage).

The following statement is from the International Atomic Energy Agency (IAEA) Web page entitled, "Pregnancy and Radiation Protection in Diagnostic Radiology:"

> "During the period of <25 weeks post conception, the central nervous system (CNS) is particularly sensitive to radiation. Fetal doses in excess of about 100 mGy [10 rad] may result in a verifiable decrease of IQ."

The following Centers for Disease Control and Prevention (CDC) guidance is from, "Radiation and Pregnancy: A Fact Sheet for Clinicians:"

> "Although radiation doses to a fetus tend to be lower than the dose to the mother, due to protection from the uterus and surrounding tissues, the human embryo and fetus are sensitive to ionizing radiation at doses greater than 0.1 gray (Gy) [10 rad]."

Reference 10, Volume II of this petition, Health Effects Associated with Radiation Exposure, is a compilation of publicly available information found on the internet describing the health effects associated with radiation exposure.

International View

The International Atomic Energy Agency (IAEA) integrated regulatory review service (IRRS) mission to the United States of America conducted in October 2010 identified a concern with the NRC's design basis accident acceptance criteria stating that:

> "Another issue related to the review and assessment is associated with determination and use of legally established criteria, in particular of the radiological acceptance criterion for design basis accidents, i.e. 250 mSv effective dose (in accordance with 10 CFR 50.67). This value is considerably higher than equivalent numbers currently used in many countries, even taking into account large conservatism embedded in demonstration of compliance with the criterion."

The IRRS report, IAEA-NS-IRRS-2010/02, contained the following suggestion related to this issue:

> S7 Suggestion: NRC should consider proper ways aimed at more direct implementation of ALARA principle in setting up the radiological acceptance criteria for design basis accidents as well as in assessment of acceptability of the results of relevant safety analysis.

The IRRS conducted a follow-up mission in February 2014, IAEA-NS-2014/01. The IRRS report on the follow-up mission identified this issue as an open item and stated the following:

> Suggestion 7: The substance of the issue and the basis for suggestion was comparably high value of the radiological acceptance criterion for design basis accidents. The validity of the original Suggestion S7 is now supported by recently published IAEA Safety Requirements for design (SSR-2/1) stating in para. 5.25. "The design shall be such that for design basis accident conditions… they have no, or only minor, radiological impacts, on or off the site, and do not necessitate any off-site intervention measures." Further on GSR Part 3, para. 1.27 states in connection with reference levels for emergency situations that "... *Any situation that resulted in a dose of greater than 100mSv being incurred acutely or in one year would be considered unacceptable*…". A number of examples from several countries were provided by the IRRS Team in which the acceptance criteria for design basis accidents in terms of the effective dose incurred within one year are in the range of 1 to 10mSv (including Bulgaria, Finland, Slovakia, UK). The 250mSv effective dose received for any 2-hour period following the onset of fission product release by any individual from the public on the boundary of the exclusion area, and the same effective dose received during the entire period of the releases on the outer boundary of the low population zone is stated in the 10 CFR 50.67. Similarly, in 10 CFR 100.11, the same effective dose is used in connection with siting for the determination of the boundary of the exclusion area and the low population zone.
>
> **Suggestion 7 is open.** Since in accordance with the Protective Action Guidelines in US the projected dose for sheltering to be initiated is 10mSv, and 250mSv used as criterion for design basis accidents does not exclude the need for off-site intervention measures.

Conclusion and Recommendations

The objective of GDC-19 appears to have been to ensure that control room operators are protected in a comfortable environment, so they are able to devote their attention to mitigating the accident condition. The original objective of the reactor siting criteria was that the occurrence of any credible accident should not create an undue hazard to the health and safety of the public. The question is could both criteria be set at a level that would not be expected to produce any adverse health effects even when considering the most dose sensitive members of the public. Aligning the accident dose design criteria for both the control room and the public to a value of 10 rem TEDE would achieve this objective. In addition, adoption of a 10 rem TEDE criterion would generate a shift in emphasis from focusing on limiting the control room dose to limiting the actual environmental releases from an accident.

The preceding excerpts from health physics experts all point to a value of 10 rem as the threshold below which adverse health effects would not be expected. As can be seen from the citations a dose criterion of 25 rem is not an appropriate design criterion for the protection of the most dose sensitive members such as pregnant women and children.

It is recommended that 10 CFR 50.67 be augmented to include a new voluntary rule entitled 10 CFR 50.67a, Accident source term, Alternative dose criteria. Suggested wording for the voluntary rule is described in Appendix B. Necessary suggested revisions to GDC-19 to accommodate the voluntary rule are also included in Appendix B. Corresponding revisions to the dose criteria specified in RG 1.183 are shown in Appendix C.

Appendix D has been included to demonstrate some of the inconsistencies in the Footnotes describing the 25 rem criterion in current regulations. It should be noted that Footnotes in § 50.34 as well as Part 52 continue to reference National Bureau of Standards (NBS) Handbook 69 dated June 5, 1959. As a credible regulatory agency, the NRC should not continue to cite an outdated reference from 1959 which is not applicable to the new TEDE criteria, conflicts with Part 20 and was only intended to be used for a once in life time accidental or emergency dose to radiation workers.

Appendix E is a representation of calculated MCA dose evaluations for plants licensed by the NRC. An examination of Appendix E shows that many plants could meet the 10 rem dose design criteria without making any changes to their dose consequence analyses.

Even a cursory examination of current health physics guidance would quickly reveal that the current accident dose design criterion of 25 rem is difficult to justify. Recommendations from the health physics community lend strong credibility to the thesis that current NRC accident dose regulatory design criterion for the control room may be unnecessarily low while the criterion for members of the public is unjustifiably high. Therefore, it is recommended that the NRC staff re-evaluate the power reactor regulatory accident dose design criteria to ensure that the resulting design of accident mitigation systems are not perceived to emphasize protection of the control room operator over protection of the public. It is hoped that the concerns and proposed changes described in this petition could serve as a point of departure for such evaluations.

PETITION FOR RULE MAKING
DESIGN BASIS ACCIDENT DOSE CRITERIA

APPENDICES:

Appendix A: The Influence of Early Reactor Siting on DBA Acceptance Criteria

Appendix B: Proposed Revisions to § 50.67 Accident source term and GDC-19

Appendix C: Proposed Revision to RG 1.183 Accident Dose Criteria

Appendix D: Comparisons of Footnotes Describing the 25 rem Dose Criterion in Current Regulations to the Original Footnote in 10 CFR 100.11

Appendix E: MCA[51] Doses for Operating Plants

[51] In guidance the MCA is termed a loss of coolant accident (LOCA). The dose consequence MCA/LOCA is evaluated using a deterministic source term assuming a substantial fuel melt with an appreciable release of fission products into the containment. The dose consequence LOCA is a separate and distinct evaluation to test the ruggedness of a plant's accident mitigation SSCs. To meet the requirements of § 50.46 Acceptance criteria for emergency core cooling systems for light-water nuclear power reactors, plants must demonstrate that the worst case break in the main coolant system will not result in the calculated maximum fuel element cladding temperature exceeding 2,200° F, which is well below the temperature for fuel melt (4,890–5,070 °F).

Background:

Hazard Summary Reports issued in the 1950's included the dose consequences from a maximum credible accident (MCA) also referred to as a maximum hypothetical accident (MHA). Such evaluations assumed that the plant experienced a substantial core melt but that the containment held at its design basis leak rate. Applicants then evaluated the radiological conditions for such an event and proffered various suggestions for dose acceptance criteria. The AEC evaluated these applications on a case by case basis without the benefit of a prescribed set of assumptions regarding the degree of core damage or defined dose acceptance criteria. TID-14844 served both purposes. It defined the degree of core damage to be assumed and described dose acceptance criteria.

As stated in Part 100, TID-14844 contains a procedural method and a sample calculation that result in distances roughly reflecting current siting practices of the Commission[52]. In other words, using the methodology contained in TID-14844, future reactors could be sited with the same relative degree of radiological safety in terms of designing for an MHA as the operating reference reactors that had been sited prior to the promulgation of Part 100. TID-14844 defines substantial core melt with specific percentages of core fission products released into the containment; 100% of the noble gases, 50 percent of the halogens and 1 percent of the solids. A plateout factor of 50% is applied to the halogen activity resulting in 25% of the halogens available for release from containment leakage. The thyroid dose from the inhalation of iodine is then calculated using a 0.1% per day containment leak rate and conservative estimates for atmospheric dispersion factors. TID-14844 calculates the whole-body dose which is compared to the 25 rem criterion based on the direct dose from unshielded containment structures with the fuel melt source term modeled as a point source. The whole-body dose from the cloud of leaked fission products was considered to be negligible as compared to the direct dose from unshielded containment structures which were common during this time period.

> *TID-14844, Assumption 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."*

TID-14844 developed EAB and LPZ distances as a function of power level based on these two considerations. Figure 1, from TID-14844 (shown on page 4 of this Appendix) shows the results for the EAB. As can be seen from an examination of Figure 1, for power levels up to approximately 300 MWt the direct shine from unshielded containments dominated the evaluation. For example, a reactor with a power level of 100 MWt would require an EAB radius of 400 meters based on a 2-hour direct gamma dose of 25 rads while the 2-hour thyroid dose criterion of 300 rem would allow for a smaller EAB radius of approximately 200 meters. Therefore, the 25 rem whole body dose criterion would control the size of the EAB for a 100 MWT reactor. In contrast, a reactor with a power level of 600 MWt would require an EAB radius of about 800 meters based on the 2-hour thyroid dose criterion of 300 rem while the 2-hour direct gamma dose of 25 rads would allow for an EAB radius of approximately 600 meters. As a

[52] TID-14844 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) can be used to calculate distance requirements for reactor sites which are generally consistent with current siting practices."

result, the 300 rem thyroid dose criterion would control the size of the EAB for a 600 MWT reactor. Therefore, for the larger reference reactors considered in TID-14844, an acceptable EAB satisfying the thyroid dose criterion required distances on the order of 0.5 miles at which the direct shine from an unshielded containment was substantially reduced.

For the larger reference reactors considered in TID-14844, the thyroid dose was the controlling factor in the determination of an acceptable EAB distance. Using the TID-14844 methodology, the significance of the 25 rem whole body dose limitation reduced as distance from the source increased resulting in the thyroid dose being the governing factor in the determination of an acceptable outer boundary for the LPZ for all power levels considered. Examination of TID-14844 methodology reveals that the significance of the 25 rem whole body reference value as an EAB determinate only applied to low power reactors with unshielded containment structures. Examination of the methodologies used in TID-14844 reveals that for larger reactors with unshielded containments and for any reactor with an effectively shielded containment, the 25 rem whole body reference value had no significance in the determination of an acceptable EAB or LPZ boundary.

The fact that the 300 rem criterion was controlling for the siting of all large power reactors cannot be over stated. The 25 rem whole body criterion had no influence on the siting of any modern power reactor.

From: TID-14844, Section 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 applied to a number of reactor projects that have been proposed or are currently authorized for construction. These results are given in Table VIII."

Reproduction of TID-14844 Table VIII. Calculated Distances for Selected Reactors

		Exclusion Area		Low Population Area	Population Center Distance	
Reactor	Power Level (MWt)	Calculated Distance (miles)	Actual Distance (miles)	Calculated Distance (miles)	Calculated Distance (miles)	Actual 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
PG&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
CVTR	60	0.22	0.50	1.6	2.1	25.0
EIK 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.

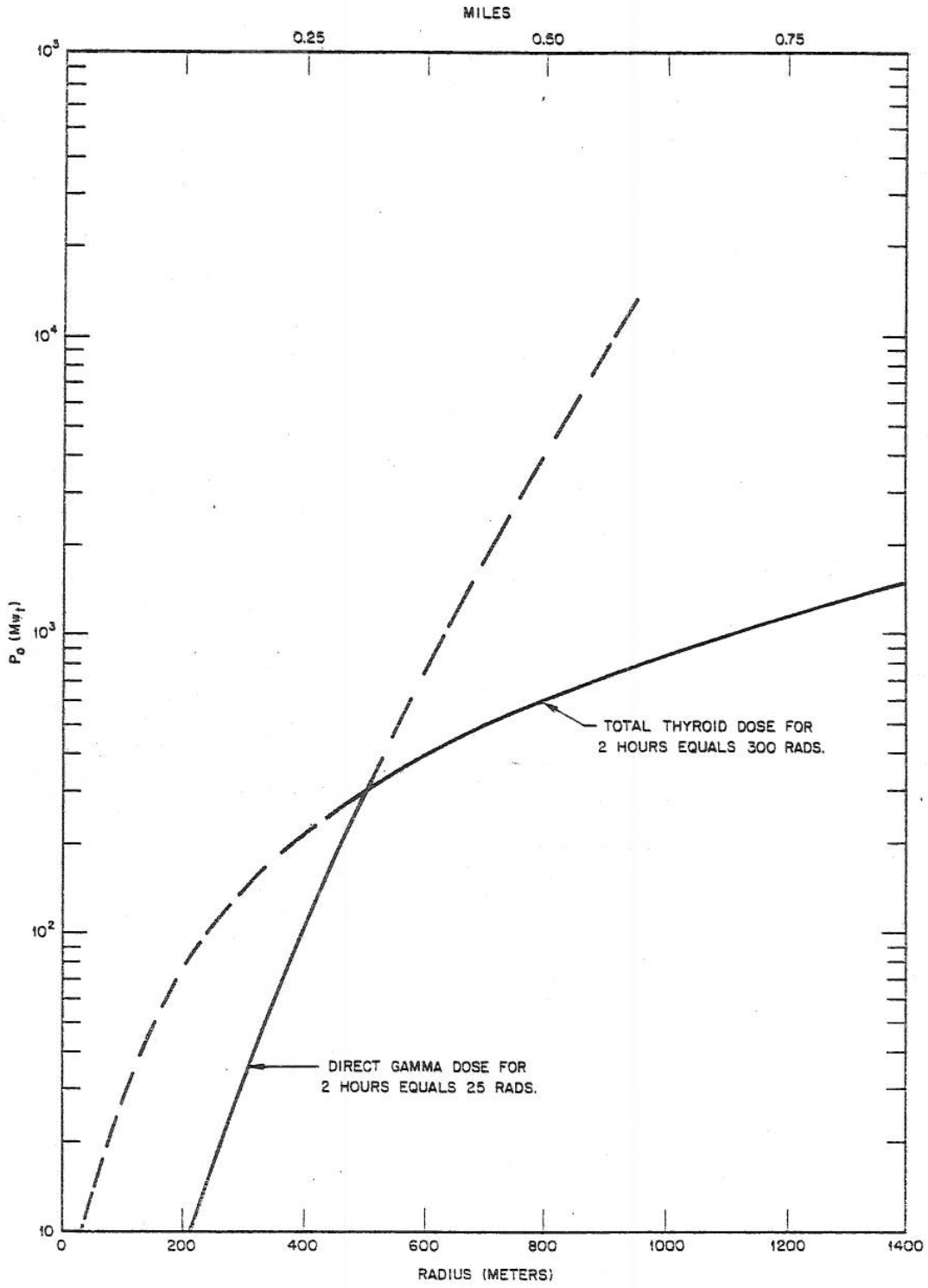

Figure 1. Exclusion Radius Determination.

TID-14844, page 32, Figure 1

Unshielded Containment Structures:

The Yankee Nuclear Power Station (YNPS)[53] received its operating license (OL) prior to the promulgation of Part 100 and served as one of the reference plants evaluated in TID-14844 (See TID-14844 Table VIII Reproduced on page 3 of this Appendix). The containment structure for the YNPS was referred to as the vapor container (VC) and consisted of a nominally one-inch thick steel sphere. The VC was elevated above ground to allow the major primary system components to be hoisted from rail cars and positioned in place within the VC using the polar crane. The concrete control room wall facing the VC was four feet thick to shield the operators in the event of a major release of fission products into the VC. After the accident at Three Mile Island Unit 2, corporate radiological engineers evaluated the dose profile for the YNPS using a TID-14844 accident source term. Results indicated that post-accident on-site dose rates could present an extreme radiological hazard. When engineers recommended that a shield wall be erected to reduce the direct shine from the VC, management asked if YNPS met Part 100 criteria. The answer of course was yes; YNPS did meet the Part 100 dose criteria, but the irony is that the plant was not designed to meet Part 100 but rather the criteria of Part 100 were selected based on the dose profile of YNPS and several other plants that had already been licensed by the Atomic Energy Commission[54] (AEC) before Part 100 was adopted (see TID-14844 Table VIII reproduced on page 3 of this Appendix)[55].

The AEC licensed several other nuclear power plants without significantly shielded containment structures including: Dresden 1, OL issued 1959; Big Rock Point, OL Issued 1964 and San Onofre 1, OL issued 1967. Under accident conditions, an unshielded containment structure represented a severe radiological hazard for plant personnel and nearby residents. The limitation of 25 rem whole body in a two-hour period was intended to limit the non-stochastic health effects caused by the direct radiation from unshielded containments thereby insuring that serious injury to individuals off-site would be avoided if an unlikely, but still credible, accident should occur. The last nuclear power plant with an unshielded containment structure was licensed in the late 1960s.

The Final Hazards Summary Reports for both the YNPS (1959) and the Dresden Nuclear Power Station (1957) included estimated direct radiation doses from their unshielded containments and referenced NBS Handbook 59 once in a life time dose for emergency workers of 25 rem as an acceptance criterion. It should be noted that the Hazards Analysis for the Consolidated Edison Reactor (Indian Point Unit I) did not reference the NBS Handbook 59 once in a life time dose for emergency workers of 25 rem as an acceptance criterion.

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[53] Yankee Rowe achieved criticality on August 19, 1960 and after extensive physics testing started commercial operation in July, 1961 at a power level of 485 MWt. On December 24, 1963 the plant was granted a license to operate the plant at the full design power level of 600 MWt.

[54]"Reminiscing almost 20 years later, regulatory staff members, who had worked on this draft, recalled trying to find a set of parameters and assumptions which would fit essentially all the previously approved reactor site combinations, within some broader, generally acceptable framework." Nuclear Reactor Safety, On the History of the Regulatory Process, David Okrent, The University of Wisconsin Press, 1981, footnote, page 39.

[55] TID-14844 Section 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 applied to a number of reactor projects that have been proposed or are currently authorized for construction. These results are given in Table VIII."

The Indian Point Unit I reactor included a concrete enclosure surrounding the steel containment vessel as described in the following quote from the Report on Hazards Analysis and Design for Containment Vessel, Consolidated Edison reactor, August 29, 1958:

VIII. RADIATION SHIELDING

> *The direct radiation shielding as mentioned previously consists of a complete concrete enclosure, surrounding the containment vessel, formed of a 5 ft-6 in. concrete wall with a 2 ft-9 in. domed concrete roof. This external shield has been designed to provide adequate radiation protection under normal operating conditions as well as to provide radiation protection in the case of a reactor incident within the containment vessel. The design has been based upon a company-imposed requirement that additional generating capacity planned for the site at Indian Point be maintained in an operable condition even if such an incident occurs. The imposition of this additional requirement further reduces the dosages below the values permissible for the public.*

The following table displays the assumptions used to evaluate the worst-case accident considered to be credible for the three major reactors that were sited prior to the promulgation of 10 CFR 100.11.

Applicants pre § 100.11 Maximum Hypothetical Accident (MHA) Source Terms

Plant	Reactor & Containment Description	Release to Containment % of core inventory			Containment Leak Rate	Applicant's Suggested Acceptance Criteria (rem)		Stated Conclusion
		Iodines	Noble Gas	Solids		External	Internal	
Dresden PHSR[56] May 1957	BWR Unshielded spherical containment partially below grade	100%	100%	30%	0.5 % / day	25 NBS Handbook 59	100 any organ	No one in the environment of the plant site would, in all probability, receive more than the maximum permissible emergency radiation exposure, provided that evacuation of the affected area is effected reasonably promptly (within, say, 8 hours) after a "worst reasonable accident".
Con. Ed. Indian Point 1 Hazards Analysis August 1958	PWR Steel Sphere 50% below grade with concrete shield enclosure building	100%	100%	10%	0.1% / day	Not found		Not found
Yankee FHSR[57] September 1959	PWR Steal Sphere elevated above grade	20%	20%	20% (Strontium)	70 ft^3/hr	25 NBS Handbook 59	15.7 Thyroid[58][59]	Not found

[56] Preliminary Hazards Summary Report
[57] Final Hazards Summary Report
[58] Additional information from Yankee FHSR September 1959: K. Z. Morgan https://archive.org/details/in.ernet.dli.2015.220469/page/n153 et al have published figures which relate activity to dose in the critical organ. For Iodine-131, he suggested a maximum permissible inhalation of 17 microcuries which would result, at 15% uptake, in 2.5 microcuries in the gland and a resultant dose of 15.7 rem in one year. In order to inhale 17 microcuries in 8 hr at the standard respiratory rate of 13 liters per minute, the air concentration must be 2.7 x 10^{-6} microcuries per milliliter.
[59] Additional information from Yankee FHSR September 1959: Kuper and Cowan https://www.worldcat.org/title/proceedings-of-the-second-united-nations-international-conference-on-the-peaceful-uses-of-atomic-energy-held-in-geneva-1-september-13-september-1958/oclc/9381034, suggested a value of 2,000 rad or 400 microcuries in the thyroid at 2 hr after the incident as a value below which no immediate injury would be expected in adults, although damage to children or delayed effects in adults is a possibility. This corresponds to 2,660 microcuries inhaled which for an 8 hr exposure calls for an air-borne concentration of 4.25 x 10-4 microcuries per milliliter. This is clearly an emergency level and is substantially higher than predicted under the most unfavorable meteorological conditions.

§ 50.67 Accident source term. [As stated in current regulations]

The proposed revision would add § 50.67a as follows:

§ 50.67a Accident source term. Alternative dose criteria.

(a) Applicability. The requirements of this section apply to all holders of operating licenses and holders of renewed licenses under part 54 of this chapter who seek to revise the dose acceptance criteria in their design basis radiological analyses.

(b) Requirements. (1) Licensees who seek to revise the dose acceptance criteria in their design basis radiological analyses shall apply for a license amendment under § 50.90. The application shall contain an evaluation of the consequences of applicable design basis accidents[1] in their safety analysis report.

(2) The NRC may issue the amendment only if the applicant's analysis demonstrates with reasonable assurance that:

(i) An individual located at any point on the boundary of the exclusion area for any 2-hour period following the onset of the postulated fission product release, would not receive a radiation dose in excess of 100 mSv (10 rem)[2] total effective dose equivalent (TEDE).

(ii) An individual located at any point on the outer boundary of the low population zone, 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 radiation dose in excess of 100 mSv (10 rem) total effective dose equivalent (TEDE).

(iii) Adequate radiation protection is provided to permit access to and occupancy of the control room under accident conditions without personnel receiving radiation exposures in excess of 100 mSv (10 rem)[3] total effective dose equivalent (TEDE) for the duration of the accident.

[1] The fission product release assumed for these calculations should be based upon a major accident, hypothesized for purposes of design analyses 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 into the containment.

[2] The use of 100 mSv (10 rem) TEDE is not intended to imply that this value constitutes an acceptable limit for emergency doses to the public under accident conditions. Rather, this 100 mSv (10 rem) TEDE value has been stated in this section as a reference value, which can be used in the evaluation of proposed design basis changes with respect to potential reactor accidents of exceedingly low probability of occurrence and low risk of public exposure to radiation. Although radiation may cause cancer at high doses and high dose rates, public health data do not absolutely establish the occurrence of cancer following exposure to low doses and dose rates below about 100 mSv (10 rem).

[3] Although radiation may cause cancer at high doses and high dose rates, public health data do not absolutely establish the occurrence of cancer following exposure to low doses and dose rates below about 100 mSv (10 rem).

Proposed revision to GDC-19:

Criterion 19—Control room. A control room shall be provided from which actions can be taken to operate the nuclear power unit safely under normal conditions and to maintain it in a safe condition under accident conditions, including loss-of-coolant accidents. Adequate radiation protection shall be provided to permit access and occupancy of the control room under accident conditions without personnel receiving radiation exposures in excess of 5 rem whole body, or its equivalent to any part of the body, for the duration of the accident. Equipment at appropriate locations outside the control room shall be provided (1) with a design capability for prompt hot shutdown of the reactor, including necessary instrumentation and controls to maintain the unit in a safe condition during hot shutdown, and (2) with a potential capability for subsequent cold shutdown of the reactor through the use of suitable procedures.

Applicants for and holders of construction permits and operating licenses under this part who apply on or after January 10, 1997, applicants for design approvals or certifications under part 52 of this chapter who apply on or after January 10, 1997, applicants for and holders of combined licenses or manufacturing licenses under part 52 of this chapter who do not reference a standard design approval or certification, or holders of operating licenses using an alternative source term under § 50.67, shall meet the requirements of this criterion, except that with regard to control room access and occupancy, adequate radiation protection shall be provided to ensure that radiation exposures shall not exceed 0.05 Sv (5 rem) total effective dose equivalent (TEDE) as defined in § 50.2 for the duration of the accident.

Holders of operating licenses using an alternative source term under § 50.67a, shall meet the requirements of this criterion, except that with regard to control room access and occupancy, adequate radiation protection shall be provided to ensure that radiation exposures shall not exceed 100 mSv (10 rem)[1] total effective dose equivalent (TEDE) as defined in § 50.2 for the duration of the accident.

[1] Although radiation may cause cancer at high doses and high dose rates, public health data do not absolutely establish the occurrence of cancer following exposure to low doses and dose rates below about 100 mSv (10 rem).

Regulatory Guide (RG) 1.183, Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors includes Table 6, "Accident Dose Criteria," which contains the accident dose criteria for the maximum credible accident (MCA), termed the loss of coolant accident (LOCA) in guidance, as well as other important design basis accidents (DBAs) which can have significant off-site doses. DBAs which have a higher probability of occurrence than the MCA have traditionally had lower dose acceptance criteria. These lower criteria were expressed as being either "well within (25%)," or "a small fraction (10%)" of the 10 CFR 100.11 or 10 CFR 50.67 dose criteria. The proposed revisions to the accident dose criteria for accidents which have a higher probability of occurrence includes revised definitions of "well within," as 50% of the revised MCA criterion, and "a small fraction," as 25% of the revised MCA criterion.

The proposed revision also includes a significant change to the existing dose criteria for the main steam line break and the steam generator tube rupture when the source term is based on either the licensee's determination of fuel damage or for the pre-incident iodine spike case. The existing criterion for these accident cases is 25 rem. The existing use of the 25 rem criterion is not appropriate for any accident except the MCA as described in the footnote in 10 CFR 50.67[60]. Therefore, the proposed revision for these accident cases includes an additional change reducing the criteria for these cases to "well within" the revised dose acceptance criteria for the MCA. The proposed revisions, expressed in Total Effective Dose Equivalent (TEDE), include the following:

- For the MCA (LOCA), the dose criterion is revised from 25 rem to 10 rem
- For the BWR Main Steam Line Break Accident with fuel damage or the pre-incident spike, the dose criterion is revised from 25 rem to 5 rem and remains 2.5 rem for the coincident iodine spike case.
- For the BWR Rod Drop Accident, the dose criterion is revised from 6.3 rem to 5 rem.
- For the PWR Steam Generator Tube Rupture with fuel damage or a pre-incident spike, the dose criterion is revised from 25 rem to 5 rem and remains 2.5 rem for the coincident iodine spike.
- For the PWR Main Steam Line Break accident with fuel damage or a pre-incident spike, the dose criteria is revised from 25 rem to 5 rem and remains 2.5 rem for the coincident iodine spike.
- For the PWR Locked Rotor Accident, the dose criterion remains 2.5 rem.
- For the PWR Rod Ejection Accident, the dose criterion is revised from 6.3 rem to 5 rem.
- For the Fuel Handling Accident, the dose criterion is revised from 6.3 rem to 5 rem.

[60] 10 CFR 50.67, Footnote 1: The fission product release assumed for these calculations should be based upon a major accident, hypothesized for purposes of design analyses 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.

Table 6a[61]
Alternative Accident Dose Criteria

Accident or Case	EAB and LPZ Dose Criteria	Analysis Release Duration
LOCA	10 rem TEDE	30 days for containment, ECCS, and MSIV (BWR) leakage
BWR Main Steam Line Break		Instantaneous puff
Fuel Damage or Pre-incident Spike	5 rem TEDE	
Equilibrium Iodine Activity	2.5 rem TEDE	
BWR Rod Drop Accident	5 rem TEDE	24 hours
PWR Steam Generator Tube Rupture		Affected SG: time to isolate; Unaffected SG(s): until cold shutdown is established
Fuel Damage or Pre-incident Spike	5 rem TEDE	
Coincident Iodine Spike	2.5 rem TEDE	
PWR Main Steam Line Break		Until cold shutdown is established
Fuel Damage or Pre-incident Spike	5 rem TEDE	
Coincident Iodine Spike	2.5 rem TEDE	
PWR Locked Rotor Accident	2.5 rem TEDE	Until cold shutdown is established
PWR Rod Ejection Accident	5 rem TEDE	30 days for containment pathway; until cold shutdown is established for secondary pathway
Fuel Handling Accident	5 rem TEDE	2 hours

[61] Table 6a provides revised dose acceptance criteria for those licensees choosing to adopt the proposed voluntary rule § 50.67a Accident source term. Alternative dose criteria.

The purpose of this appendix is to display the inconsistencies in the Footnotes describing the 25 rem criterion in current regulations. In addition, the Footnotes describing the 25 rem criterion in § 50.34 and Part 52 continue to reference National Bureau of Standards (NBS) Handbook 69 dated June 5, 1959. As a credible regulatory agency, the NRC should not continue to cite an outdated reference from 1959 which is not applicable to the new TEDE criteria, conflicts with Part 20 and was only intended to be used for a once in life time accidental or emergency dose to radiation workers.

The footnote describing the use of 25 rem first appeared in 10 CFR 100.11 is shown below:

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 produce release[1] 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) 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[2] or a total radiation dose in excess of 300 rem[2] 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, the boundary of the population center shall be determined upon consideration of population distribution. Political boundaries are not controlling in the application of this guide. 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.

[27 FR 3509, Apr. 12, 1962, as amended at 31 FR 4670, Mar. 19, 1966; 38 FR 1273, Jan. 11, 1973; 40 FR 8793, Mar. 3, 1975; 40 FR 26527, June 24, 1975; 53 FR 43422, Oct. 27, 1988; 64 FR 48955, Sept. 9, 1999; 67 FR 67101, Nov. 4, 2002]

[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 once in a lifetime accidental or emergency dose for radiation workers which, according to NCRP recommendations may be disregarded in the determination of their radiation exposure status (see NBS Handbook 69 dated June 5, 1959). 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, this 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.

The footnote describing the use of 25 rem TEDE in § 50.34 is shown below:

§ 50.34 Contents of applications; technical information.

(a) Preliminary safety analysis report. Each application for a construction permit shall include a preliminary safety analysis report. The minimum information[5] to be included shall consist of the following:

(1) Stationary power reactor applicants for a construction permit who apply on or after January 10, 1997, shall comply with paragraph (a)(1)(ii) of this section. All other applicants for a construction permit shall comply with paragraph (a)(1)(i) of this section.

(i) A description and safety assessment of the site on which the facility is to be located, with appropriate attention to features affecting facility design. Special attention should be directed to the site evaluation factors identified in part 100 of this chapter. The assessment must contain an analysis and evaluation of the major structures, systems and components of the facility which bear significantly on the acceptability of the site under the site evaluation factors identified in part 100 of this chapter, assuming that the facility will be operated at the ultimate power level which is contemplated by the applicant. With respect to operation at the projected initial power level, the applicant is required to submit information prescribed in paragraphs (a)(2) through (a)(8) of this section, as well as the information required by this paragraph, in support of the application for a construction permit, or a design approval.

(ii) A description and safety assessment of the site and a safety assessment of the facility. It is expected that reactors will reflect through their design, construction and operation an extremely low probability for accidents that could result in the release of significant quantities of radioactive fission products. The following power reactor design characteristics and proposed operation will be taken into consideration by the Commission:

(A) Intended use of the reactor including the proposed maximum power level and the nature and inventory of contained radioactive materials;

(B) The extent to which generally accepted engineering standards are applied to the design of the reactor;

(C) The extent to which the reactor incorporates unique, unusual or enhanced safety features having a significant bearing on the probability or consequences of accidental release of radioactive materials;

(D) 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. Special attention must be directed to plant design features intended to mitigate the radiological consequences of accidents. In performing this assessment, an applicant shall assume a fission product release[6] from the core into the containment assuming that the facility is operated at the ultimate power level contemplated. The applicant shall perform an evaluation and analysis of the postulated fission product release, using the expected demonstrable containment leak rate and any fission product cleanup systems intended to mitigate the consequences of the accidents, together with applicable site characteristics, including site meteorology, to evaluate the offsite radiological consequences. Site characteristics must comply with part 100 of this chapter. The evaluation must determine that:

(1) An individual located at any point on the boundary of the exclusion area for any 2 hour period following the onset of the postulated fission product release, would not receive a radiation dose in excess of 25 rem[7] total effective dose equivalent (TEDE).

(2) An individual located at any point on the outer boundary of the low population zone, 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 radiation dose in excess of 25 rem total effective dose equivalent (TEDE);

[5]The applicant may provide information required by this paragraph in the form of a discussion, with specific references, of similarities to and differences from, facilities of similar design for which applications have previously been filed with the Commission.

[6]The fission product release assumed for this evaluation should be based upon a major accident, hypothesized for purposes of site analysis or postulated from considerations of possible accidental events. Such accidents have generally been assumed to result in substantial meltdown of the core with subsequent release into the containment of appreciable quantities of fission products.

[7]A whole body dose of 25 rem has been stated to correspond numerically to the once in a lifetime accidental or emergency dose for radiation workers which, according to NCRP recommendations at the time could be disregarded in the determination of their radiation exposure status (see NBS Handbook 69 dated June 5, 1959). However, its use is not intended to imply that this number constitutes an acceptable limit for an emergency dose to the public under accident conditions. Rather, this dose value has been set forth in this section as a reference value, which can be used in the evaluation of plant design features with respect to postulated reactor accidents, in order to assure that such designs provide assurance of low risk of public exposure to radiation, in the event of such accidents.

Footnote 7 in 10 CFR 50.34 describes a whole body dose of 25 rem even though 10 CFR 50.34 is written for 25 rem TEDE; the reference to 300 rem thyroid dose has been eliminated; “the evaluation of reactors sites” has been replaced with “the evaluation of plant design features” and the phrase “of exceedingly low probability of occurrence,” has been eliminated.

Comparison of 10 CFR 50.34 footnote 7 to 10 CFR 100.11 footnote 2:

~~The~~ A whole body dose of 25 rem ~~referred~~ has been stated to ~~above corresponds~~ correspond numerically to the once in a lifetime accidental or emergency dose for radiation workers which, according to NCRP recommendations ~~may~~ at the time could be disregarded in the determination of their radiation exposure status (see NBS Handbook 69 dated June 5, 1959). However, ~~neither~~ its use ~~nor that of the 300 rem value for thyroid exposure as set forth in these site criteria guides are~~ is not intended to imply that ~~these numbers constitute~~ this number constitutes an acceptable ~~limits~~ limit for an emergency ~~doses~~ dose to the public under accident conditions. Rather, this ~~25 rem whole body value and the 300 rem thyroid value have~~ dose value has been set forth in ~~these guides~~ this section as a reference ~~values~~ value, which can be used in the evaluation of ~~reactor sites~~ plant design features with respect to ~~potential~~ postulated reactor accidents ~~of exceedingly low probability of occurrence, and~~, in order to assure that such designs provide assurance of low risk of public exposure to radiation, in the event of such accidents.

The footnote describing the use of 25 rem TEDE in § 50.67 is shown below:

§ 50.67 Accident source term.

(a) Applicability. The requirements of this section apply to all holders of operating licenses issued prior to January 10, 1997, and holders of renewed licenses under part 54 of this chapter whose initial operating license was issued prior to January 10, 1997, who seek to revise the current accident source term used in their design basis radiological analyses.

(b) Requirements. (1) A licensee who seeks to revise its current accident source term in design basis radiological consequence analyses shall apply for a license amendment under § 50.90. The application shall contain an evaluation of the consequences of applicable design basis accidents[1] previously analyzed in the safety analysis report.

(2) The NRC may issue the amendment only if the applicant's analysis demonstrates with reasonable assurance that:

(i) An individual located at any point on the boundary of the exclusion area for any 2-hour period following the onset of the postulated fission product release, would not receive a radiation dose in excess of 0.25 Sv (25 rem)[2] total effective dose equivalent (TEDE).

(ii) An individual located at any point on the outer boundary of the low population zone, 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 radiation dose in excess of 0.25 Sv (25 rem) total effective dose equivalent (TEDE).

(iii) Adequate radiation protection is provided to permit access to and occupancy of the control room under accident conditions without personnel receiving radiation exposures in excess of 0.05 Sv (5 rem) total effective dose equivalent (TEDE) for the duration of the accident.

[64 FR 72001, Dec. 23, 1999]

[1] The fission product release assumed for these calculations should be based upon a major accident, hypothesized for purposes of design analyses 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 use of 0.25 Sv (25 rem) TEDE is not intended to imply that this value constitutes an acceptable limit for emergency doses to the public under accident conditions. Rather, this 0.25 Sv (25 rem) TEDE value has been stated in this section as a reference value, which can be used in the evaluation of proposed design basis changes with respect to potential reactor accidents of exceedingly low probability of occurrence and low risk of public exposure to radiation.

10 CFR 50.67 footnote 2 is similar to 10 CFR 50.34 footnote 7 in that it refers to the evaluation of design changes as opposed to the evaluation of reactor sites. No reference to NBS Handbook 69 or to a whole body dose. The dose is expressed in both units of Sievert and rem. 10 CFR 50.67 footnote 2 retained the phrase "of exceedingly low probability of occurrence" from the original footnote in 10 CFR 100.11.

Comparison of 10 CFR 50.67 footnote 2 to 10 CFR 50.34 footnote 7:

~~A whole body dose~~ The use of 0.25 Sv (25 rem ~~has been stated to correspond numerically to the once in a lifetime accidental or emergency dose for radiation workers which, according to NCRP recommendations at the time could be disregarded in the determination of their radiation exposure status (see NBS Handbook 69 dated June 5, 1959). However, its use~~) TEDE is not intended to imply that this ~~number~~ value constitutes an acceptable limit for ~~an~~ emergency ~~dose~~ doses to the public under accident conditions. Rather, this ~~dose~~ 0.25 Sv (25 rem) TEDE value has been ~~set forth~~ stated in this section as a reference value, which can be used in the evaluation of ~~plant~~ proposed design ~~features~~ basis changes with respect to ~~postulated~~ potential reactor accidents~~, in order to assure that such designs provide assurance~~ of exceedingly low probability of occurrence and low risk of public exposure to radiation~~, in the event of such accidents~~.

Comparison of 10 CFR 50.67 footnote 2 to 10 CFR 100.11 footnote 2:

The ~~whole body dose of 25 rem referred to above corresponds numerically to the once in a lifetime accidental or emergency dose for radiation workers which, according to NCRP recommendations may be disregarded in the determination of their radiation exposure status (see NBS Handbook 69 dated June 5, 1959). However, neither its~~ use ~~nor that of the 300 rem value for thyroid exposure as set forth in these site criteria guides are~~ of 0.25 Sv (25 rem) TEDE is not intended to imply that ~~these numbers constitute~~ this value constitutes an acceptable ~~limits~~ limit for emergency doses to the public under accident conditions. Rather, this 0.25 Sv (25 rem ~~whole body~~) TEDE value ~~and the 300 rem thyroid value have~~ has been ~~set forth~~ stated in ~~these guides~~ this section as a reference ~~values~~ value, which can be used in the evaluation of ~~reactor sites~~ proposed design basis changes with respect to potential reactor accidents of exceedingly low probability of occurrence, and low risk of public exposure to radiation.

The footnotes describing the use of 25 rem TEDE in Part 52 are shown below:

Subpart A Early Site Permits 52.17 Contents of applications; technical information.

§ 52.17 Contents of applications; technical information.

(a) For applications submitted before September 27, 2007, the rule provisions in effect at the date of docketing apply unless otherwise requested by the applicant in writing. The application must contain:

(1) A site safety analysis report. The site safety analysis report shall include the following:

(i) The specific number, type, and thermal power level of the facilities, or range of possible facilities, for which the site may be used;

(ii) The anticipated maximum levels of radiological and thermal effluents each facility will produce;

(iii) The type of cooling systems, intakes, and outflows that may be associated with each facility;

(iv) The boundaries of the site;

(v) The proposed general location of each facility on the site;

(vi) The seismic, meteorological, hydrologic, and geologic characteristics of the proposed site with appropriate consideration of the most severe of the natural phenomena that have been historically reported for the site and surrounding area and with sufficient margin for the limited accuracy, quantity, and period of time in which the historical data have been accumulated;

(vii) The location and description of any nearby industrial, military, or transportation facilities and routes;

(viii) The existing and projected future population profile of the area surrounding the site;

(ix) A description and safety assessment of the site on which a facility is to be located. The assessment must contain an analysis and evaluation of the major structures, systems, and components of the facility that bear significantly on the acceptability of the site under the radiological consequence evaluation factors identified in paragraphs (a)(1)(ix)(A) and (a)(1)(ix)(B) of this section. In performing this assessment, an applicant shall assume a fission product release[1] from the core into the containment assuming that the facility is operated at the ultimate power level contemplated. The applicant shall perform an evaluation and analysis of the postulated fission product release, using the expected demonstrable containment leak rate and any fission product cleanup systems intended to mitigate the consequences of the accidents, together with applicable site characteristics, including site meteorology, to evaluate the offsite radiological consequences. Site characteristics must comply with part 100 of this chapter. The evaluation must determine that:

(A) An individual located at any point on the boundary of the exclusion area for any 2 hour period following the onset of the postulated fission product release, would not receive a radiation dose in excess of 25 rem[2] total effective dose equivalent (TEDE).

(B) An individual located at any point on the outer boundary of the low population zone, 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 radiation dose in excess of 25 rem TEDE;

[1]The fission product release assumed for this evaluation should be based upon a major accident, hypothesized for purposes of site analysis or postulated from considerations of possible accidental events. Such accidents have generally been assumed to result in substantial meltdown of the core with subsequent release into the containment of appreciable quantities of fission products.

[2] A whole body dose of 25 rem has been stated to correspond numerically to the once in a lifetime accidental or emergency dose for radiation workers which, according to NCRP recommendations at the time could be disregarded in the determination of their radiation exposure status (see NBS Handbook 69 dated June 5, 1959). However, its use is not intended to imply that this number constitutes an acceptable limit for an emergency dose to the public under accident conditions. Rather, this dose value has been set forth in this section as a reference value, which can be used in the evaluation of plant design features with respect to postulated reactor accidents, to assure that these designs provide assurance of low risk of public exposure to radiation, in the event of an accidents.[62]

As shown below, the footnotes in Part 52 regarding the 25 rem value are very close to the footnote in 10 CFR 50.34.

Comparison of 10 CFR 52.17 footnote 2 to 10 CFR 50.34 footnote 7:

A whole body dose of 25 rem has been stated to correspond numerically to the once in a lifetime accidental or emergency dose for radiation workers which, according to NCRP recommendations at the time could be disregarded in the determination of their radiation exposure status (see NBS Handbook 69 dated June 5, 1959). However, its use is not intended to imply that this number constitutes an acceptable limit for an emergency dose to the public under accident conditions. Rather, this dose value has been set forth in this section as a reference value, which can be used in the evaluation of plant design features with respect to postulated reactor accidents, ~~in order~~ to assure that ~~such~~ these designs provide assurance of low risk of public exposure to radiation, in the event of ~~such~~ an accidents.

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[62] There is a tense error in this footnote. Subsequent footnotes in Part 52 concerning the 25 rem dose criterion correct this error.

With the exception of the correction of the grammatical error in the phrase “in the event of an accidents” to “in the event of an accident,” the remaining footnotes in Part 52 are consistent with the footnote in 10 CFR 52.17 as shown in the excerpts below:

Subpart B – Standard Design Certifications
52.47 Contents of applications; technical information. Footnote 4

A whole body dose of 25 rem has been stated to correspond numerically to the once in a lifetime accidental or emergency dose for radiation workers which, according to NCRP recommendations at the time could be disregarded in the determination of their radiation exposure status (see NBS Handbook 69 dated June 5, 1959). However, its use is not intended to imply that this number constitutes an acceptable limit for an emergency dose to the public under accident conditions. This dose value has been set forth in this section as a reference value, which can be used in the evaluation of plant design features with respect to postulated reactor accidents, to assure that these designs provide assurance of low risk of public exposure to radiation, in the event of an accident.

Subpart C—Combined Licenses
52.79 Contents of applications; technical information in final safety analysis report. Footnote 6

A whole body dose of 25 rem has been stated to correspond numerically to the once in a lifetime accidental or emergency dose for radiation workers which, according to NCRP recommendations at the time could be disregarded in the determination of their radiation exposure status (see NBS Handbook 69 dated June 5, 1959). However, its use is not intended to imply that this number constitutes an acceptable limit for an emergency dose to the public under accident conditions. Rather, this dose value has been set forth in this section as a reference value, which can be used in the evaluation of plant design features with respect to postulated reactor accidents, to assure that these designs provide assurance of low risk of public exposure to radiation, in the event of an accident.

Subpart E—Standard Design Approvals
52.137 Contents of applications; technical information. Footnote 10

A whole body dose of 25 rem has been stated to correspond numerically to the once in a lifetime accidental or emergency dose for radiation workers which, according to NCRP recommendations at the time could be disregarded in the determination of their radiation exposure status (see NBS Handbook 69 dated June 5, 1959). However, its use is not intended to imply that this number constitutes an acceptable limit for an emergency dose to the public under accident conditions. Rather, this dose value has been set forth in this section as a reference value, which can be used in the evaluation of plant design features with respect to postulated reactor accidents, to assure that these designs provide assurance of low risk of public exposure to radiation, in the event of an accident.

Subpart F—Manufacturing Licenses
52.157 Contents of applications; technical information in final safety analysis report.
Footnote 12:

A whole body dose of 25 rem has been stated to correspond numerically to the once in a lifetime accidental or emergency dose for radiation workers which, according to NCRP recommendations at the time could be disregarded in the determination of their radiation exposure status (see NBS Handbook 69 dated June 5, 1959). However, its use is not intended to imply that this number constitutes an acceptable limit for an emergency dose to the public under accident conditions. Rather, this dose value has been set forth in this section as a reference value, which can be used in the evaluation of plant design features with respect to postulated reactor accidents, to assure that these designs provide assurance of low risk of public exposure to radiation, in the event of an accident.

The table displays typical MCA/LOCA doses for plants licensed by the NRC. Dose values are shown in TEDE for those plants that have adopted the AST under 10 CFR 50.67. For plants that use the source term from TID-14844 the results are shown as Whole Body/Thyroid. The bolded italic values shown in red would exceed the proposed 10 rem criterion using existing calculation assumptions. This representation suggests that many of the current operating reactors could meet a uniform 10 rem dose design criteria without making any changes to their dose consequence analyses. Plant names and types have been deleted as this information is not pertinent to this petition.

		AST LOCA Dose			**TID 14844 LOCA Dose**		
Plant	**Type**	**TEDE (rem)**			**Whole Body/Thyroid (rem)**		
Deleted	Deleted	**EAB**	**LPZ**	**CR**	**EAB**	**LPZ**	**CR**
Deleted	Deleted	***10.49***	***2.56***	***3.77***			
Deleted	Deleted	9.08	0.673	2.06			
Deleted	Deleted	***16.5***	***3.0***	***2.5***			
Deleted	Deleted	***16.5***	***3.0***	***2.5***			
Deleted	Deleted	***12.2***	***2.99***	***4***			
Deleted	Deleted	***12.2***	***2.99***	***4***			
Deleted	Deleted	1.02	1.25	1.25			
Deleted	Deleted	1.02	1.25	1.25			
Deleted	Deleted	1.02	1.25	1.25			
Deleted	Deleted	0.64	1.36	3.62			
Deleted	Deleted	0.64	1.36	3.62			
Deleted	Deleted	***12.2***	***2.99***	***4***			
Deleted	Deleted	***12.2***	***2.99***	***4***			
Deleted	Deleted				4.8/130	1.3/130	0.45/26
Deleted	Deleted	1.85	0.46	4.57			
Deleted	Deleted	1.85	0.46	4.57			
Deleted	Deleted	5.55	3.19	2.21			
Deleted	Deleted	5.55	3.19	2.21			
Deleted	Deleted	***17.11***	***7.28***	***4.7***			
Deleted	Deleted	4.1	4	3.5			
Deleted	Deleted				0.7/59	0.25/44	1.2/40
Deleted	Deleted				0.7/59	0.25/44	1.2/40
Deleted	Deleted	1.0	5.6	3.2			
Deleted	Deleted				3.5/234	0.4/25	1.1/19
Deleted	Deleted	5.6	1	4.44			
Deleted	Deleted	5.6	1	4.44			
Deleted	Deleted	***21.48***	***8.3***	***4.56***			
Deleted	Deleted	***21.48***	***8.3***	***4.56***			

		AST LOCA Dose			TID 14844 LOCA Dose		
Plant	**Type**	**TEDE (rem)**			**Whole Body/Thyroid (rem)**		
Deleted	Deleted	**EAB**	**LPZ**	**CR**	**EAB**	**LPZ**	**CR**
Deleted	Deleted	1.58	0.868	4.87			
Deleted	Deleted	1.58	0.868	4.87			
Deleted	Deleted	0.25	0.6	4.2			
Deleted	Deleted	0.34	0.75	4.9			
Deleted	Deleted	0.34	0.75	4.9			
Deleted	Deleted	4.38	1.72	3.93			
Deleted	Deleted	9	3.8	3.7			
Deleted	Deleted	***17.8***	***1.3***	***3.99***			
Deleted	Deleted	2.91	0.69	4.2			
Deleted	Deleted	10	5.85	1.31			
Deleted	Deleted	***19.6***	***11.4***	***4.7***			
Deleted	Deleted				2.34/62.2	1.89/68.7	0.01/11.2
Deleted	Deleted	***13.2***	***6.0***	***4.9***			
Deleted	Deleted	***13.2***	***6.0***	***4.9***			
Deleted	Deleted	2.24	0.26	4.23			
Deleted	Deleted	2.24	0.26	4.23			
Deleted	Deleted	0.9	1.25	4.01			
Deleted	Deleted	0.9	1.25	4.01			
Deleted	Deleted	9.5	1.9	4.3			
Deleted	Deleted	9.5	1.9	4.3			
Deleted	Deleted	2.9	1.7	3.0			
Deleted	Deleted	9.1	4.5	3.4			
Deleted	Deleted	1.31	1.72	3.4			
Deleted	Deleted	9.02	1.6	4.81			
Deleted	Deleted	0.657	0.769	1.65			
Deleted	Deleted	1.85	0.12	2.77			
Deleted	Deleted	1.85	0.12	2.77			
Deleted	Deleted	***11.8***	***3.3***	***4.4***			
Deleted	Deleted	***11.8***	***3.3***	***4.4***			
Deleted	Deleted	***11.8***	***3.3***	***4.4***			
Deleted	Deleted	1.91	0.59	4.63			
Deleted	Deleted	***13***	***3.3***	***4***			
Deleted	Deleted				4/60	3/160	2/10
Deleted	Deleted				4/60	3/160	2/10

		AST LOCA Dose			TID 14844 LOCA Dose		
		TEDE (rem)			Whole Body/Thyroid (rem)		
Plant	**Type**	**EAB**	**LPZ**	**CR**	**EAB**	**LPZ**	**CR**
Deleted	Deleted				4/60	3/160	2/10
Deleted	Deleted	***11***	***9***	***4.7***			
Deleted	Deleted	***11***	***9***	***4.7***			
Deleted	Deleted	***20.6***	***9.8***	***4.1***			
Deleted	Deleted				2.9/98	0.7/13	0.71/5.1
Deleted	Deleted	***14.2***	***1.6***	***4.9***			
Deleted	Deleted	***14.2***	***1.6***	***4.9***			
Deleted	Deleted	2.58	2.42	4.52			
Deleted	Deleted	2.58	2.42	4.52			
Deleted	Deleted	8.47	2.63	4.08			
Deleted	Deleted	8.47	2.63	4.08			
Deleted	Deleted	2.69	1.02	4.3			
Deleted	Deleted	***15.24***	***7.67***	***3.45***			
Deleted	Deleted	4.08	1.35	4.17			
Deleted	Deleted	4.08	1.35	4.17			
Deleted	Deleted	4.4	3.38	4.73			
Deleted	Deleted				7.99/55	1.53/8.4	1.2/1.5
Deleted	Deleted				7.99/55	1.53/8.4	1.2/1.5
Deleted	Deleted	7.9	5.4	3.1			
Deleted	Deleted	5.6	2.8	3.7			
Deleted	Deleted	5.6	2.8	3.7			
Deleted	Deleted	1.1	2.5	4.7			
Deleted	Deleted	1.2	2.6	4.5			
Deleted	Deleted	***24.01***	***3.57***	***4.86***			
Deleted	Deleted	***24.01***	***3.57***	***4.86***			
Deleted	Deleted	7.8	3.8	4.8			
Deleted	Deleted	7.8	3.8	4.8			
Deleted	Deleted	***24.4***	***7.72***	***4.75***			
Deleted	Deleted	5.85	1.58	4.87			
Deleted	Deleted	5.85	1.58	4.87			
Deleted	Deleted	4.87	0.54	1.01			
Deleted	Deleted				2/85	1.5/124	0.7/30
Deleted	Deleted				2/85	1.5/124	0.7/30
Deleted	Deleted	5.3	2.37	3.93			
Deleted	Deleted				2/37	1.4/11	0.8/3.6
Deleted	Deleted						1.1/3.8
Deleted	Deleted	***9.0***	***14.0***	***3.7***			

As can be seen by the tabular values listed[63] many plants have control room doses that are very close to the regulatory design criteria. If conditions in a plant change requiring a revision to the dose consequence analysis of record, having very little margin to the regulation can result in a licensee having to submit a license amendment request for an insignificant increase in the calculated control room dose. Following the guidance governing § 50.59, if an increase in the calculated dose exceeds 10 percent of the difference between the current licensing basis value and the regulatory criterion, a licensee must submit the revised evaluation for NRC approval. For example, for a plant with a calculated control room dose of 4.75 rem the margin to the regulation would be 250 mrem. Therefore, if a change in the inputs to the calculation resulted in an increase of more than 25 mrem the licensee would be required to submit a license amendment request for NRC approval. In the opinion of the petitioned this situation represents an unnecessary regulatory burden without a commensurate increase in safety.

[63] The values displayed are representative in nature and may not reflect values in any plants current licensing basis analyses of record.

VOLUME II: PETITION FOR RULEMAKING
DESIGN BASIS ACCIDENT DOSE CRITERIA

REFERENCES:

1. Advisory Committee on Reactor Safeguards, Minutes of the Environmental Subcommittee, February 18, 1959, ML021750385.

2. Atomic Energy Commission, 10 CFR Chapter 1, Power and Test Reactors, Notice of Proposed Rule Making [on Reactor Site Criteria], (24 FRN 4184 1959), May 23, 1959.

3. Advisory Committee on Reactor Safeguards, Minutes of the Environmental Subcommittee, August 23, 1960, ML021750500.

4. Atomic Energy Commission, Report to General Manager by the Director, Division of Licensing & Regulations, Reactor Site Criteria. This report contains an important ACRS letter dated October 22, 1960. This important letter can be found on pages 21 -25 of Reference 4. The entire report is available in Adams with a Document Date May 25, 1959, ML021960199.

5. Atomic Energy Commission, 10 CFR Part 100 Reactor Site Criteria, Notice of Proposed Guides, (26 FRN 1224 1961), February 11, 1961.

6. Atomic Energy Commission, Title 10 Atomic Energy, Chapter I, Atomic Energy Commission, Part 100, Reactor Site Criteria, (27 FRN 3509 1962), April 13, 1962.

7. Relevant FRN excerpts discussing the conversion of the §100.11 criteria to 25 rem TEDE.

8. Raymond A Crandall, Petition for Rulemaking to U.S. NRC, PRM-50-87, to revise 10 CFR 50, Appendix A, "General Design Criteria for Nuclear Power Plants" and 10 CFR 50.67, "Accident Source Term," to eliminate Control Room dose criteria. Docketed May 25, 2007.

9. U.S. nuclear Regulatory Commission, 10 CFR Part 50, NRC-2007-0016; PRM-50-97, Raymond A. Crandall; Denial of Petition for Rulemaking, Federal Register Vol. 7415 January 26, 2009.

10. Health Effects Associated with Radiation Exposure. This reference is a compilation of the health effects associated with exposure to radiation from the Environmental Protection Agency, The Centers for Disease Control and Prevention and the Radiation Safety Division of Duke University and Duke Medicine.

11. Excerpts from, A Brief History of Radiation - Protection Standards, William C. Inkret, Charles B. Meinhold, and John C. Taschner, *Los Alamos Science,* Number 23 1995. Available at https://permalink.lanl.gov › object › lareport › LA-UR-95-4005-04.
    This reference shows how recommendations for safe levels of radiation exposure have been reduced as scientists learned more about the health effects of radiation.

Draft No. 1
JBG
2/26/59

Subject: MINUTES OF THE ACRS ENVIRONMENTAL SUBCOMMITTEE MEETING HELD FEBRUARY 18, 1959, IN WASHINGTON, D. C.

The ACRS Environmental Subcommittee met at 10:00 a.m., on Wednesday, February 18, 1959, in Washington, D. C.

Attendance: ACRS Subcommittee

C. R. McCullough
K. R. Osborn
L. Silverman
C. R. Williams
F. A. Gifford

J. B. Graham, Exec. Assistant

BNL

Kenneth Downes
Irving Singer

A background summary on site criteria prepared by Dr. Gifford (attached as Appendix A) was distributed to and read by the participants of the meeting.

Mr. Downes stated that, in the studies he had seen, the effect of an adequate warning system (civil defense)and the advantage of decontamination factors readily obtainable in the field were generally neglected. Dr. Silverman observed that the most recent information on D.Fs. was as follows:

MEDIA	REMOVAL EFFICIENCY (%)*
Turkish towel (2 thicknesses)	95
Handkerchief	80
Gas Mask	99.98
Respirator	90

*For 1 micron particles (optimum for lung retention)

F/11

Efficiencies for gas masks vary widely and are closely related to how well the facepiece fits the subject. Experiments with respirators in some cases show as high as 99 per cent removal and others as low as 50 per cent (Hanford data). Dr. Williams suggested that since one could not always depend upon successful execution of an evacuation plan the gains therefrom should be looked upon as gravy. Dr. Silverman reviewed the use of weighting factors (presented earlier) which apply to various aspects of the reactor and permit a quantitative approach to the making of the judgment upon the acceptability of the reactor - site combination.

Dr. McCullough stated that it was his understanding that Mr. Price might publish site criteria for public comment in late March or early April. There is a need to publish some "numbers" since operation without numbers, as has been done in the past, implies acting in an arbitrary manner.

Dr. McCullough said that the Committee must be certain that there is agreement on the philosophy which lies behind the choice of the numbers as well as the numbers themselves. In addition, one must decide on whether to state a minimum value (e.g. exclusion radius) for the best situation which must be increased in certain cases or a maximum requirement for the worst case which may be relaxed to a given degree based upon the merits of the particular case. Dr. Williams stated that on the basis of zoning experience if one sets a minimum the tendency is to reduce that minimum. He suggested stating the larger value and allowing reduction in special cases.

In the parts that follow the underscored portions were presented as an outline by Dr. McCullough. The portions not underscored were explanatory remarks or summaries of the discussion which followed.

I. Damage to Humans

A. Employees 5 rem/yr. This applies for the period from 18 years of age to 48 years of age. It was agreed that for the moment this problem would be deferred. Some hold that the term employees should apply only to those immediately concerned with the operation of the reactor.

B. Public

1. Vicinity .5 rem/yr

2. Total Population .05 rem/yr.

II. Damage to Property (crops, soil, etc.) If shortage of farmland exists this could be a problem. Not so for United States. It was agreed that this problem should be deferred.

Dr. McCullough differentiated normal vs. abnormal operation. In the course of normal operations one can expect cladding failures, stuffing box leaks, bearing failures, etc. These may result in release of some radioactivity and Part 20 should govern. Part 20 does not however apply to the abnormal operation (accident) brought about by cracking of a pump casing, rupture of high pressure piping, etc. It is not possible therefore to define an acceptable emergency dose since one cannot predict the accident. The concept of 25R as an acceptable emergency dose is not valid. It is valid, of course, under the concept for which it was initially defined.

This was in connection with the willingness to expose an individual to a dose, which could be fairly accurately estimated in advance, in order to save life or valuable property. Further, one should have interest beyond the exposure to an individual at the site boundary. What doses are seen as a function of distance beyond the site boundary and how many people are exposed?

III. Degree of Protection

A. Definite time Period - 10 years. Description of events, effects, probabilities, etc., are more realistic if one considers a period of time such as 10 years. For example, in the operation of 100 large reactors such as Dresden over a 10-year period several fuel elements may burn releasing their activity. One can design against all reasonable accidents but one can neither predict them nor insure that there can never be one.

B. Exposure Limit

1. Whole body gamma dose)
2. Internal dose) For entire Atomic Energy program* - 200,000 Roentgen Units**

The application to site criteria follows. (Assume principal portions to be underlined.)

A formula is derived which permits comparison of sites for the case where power level is the only variable. All reactors are assumed to have fission product inventories resulting from 200 days of continuous

* excluding the Isotopes program

** exposure to an individual in excess of 1000 rem is counted as 1000.

operation. Each reactor is apportioned its fair share as follows:

$$\frac{\text{Thermal power (Mw) x 10 years}}{\text{U. S. Capacity for 10-year period}} \times 200{,}000 \text{ R.U.}$$

$$= \frac{\text{MW(th) x 10}}{5{,}000 \times 10} \times 200{,}000 \text{ R.U.}$$

or 40 Roentgen Units per thermal MW

One can compute a maximum accident and an "average" accident. An arbitrary release fraction (not without basis) of .003 is assumed for the purpose of the calculation. This is the fraction of the fission product inventory which is assumed to be released outside the container.

Thus

40 MW (th) = (.003) f (population density, wind rose)

The population density distribution and the wind rose are unique to the site.

Another factor to be considered, F, is a function of the type of reactor (T) and its use (U).

$$F = F(T,U)$$

For example one could assume PWR as a base at F = .1 and assign other F values as follows:

PWR	F = .1
Dresden	F = .5
Testing	F = 5
Experimental	F = 10

Adding a containment factor (C) the generalized formula can be written:

$$\text{MW (th)}\ F\ C\ R \left[f\left(\rho_{x,y} \right)^{\overrightarrow{U}}_{x,y,z} \right] \geq 40 \text{ R.U.}$$

where

F = factor for type and use
C = containment factor
R = release fraction

ρ = population density distribution in x, y plane

$\vec{u}$ = wind velocity vector in x, y, z coordinate system

Based upon population density figures given them, Singer and Downes computed

12,000 R.U. for PWR and

326 R.U. for VBWR

[Fair share is 9,600 for PWR and 2,000 for VBWR (assumed power of 50 MW)]

IV. Kinds of Accidents

1. Nuclear runaway
2. Local overheating
   a. Deposits on fuel elements (crud)
   b. Flow stoppage in channels
   c. Flux peaking
   d. Oscillations
3. Loss of cooling
4. Chemical reaction
   a. Inside reactor
   b. Outside reactor
5. Dependent sequences

V. Containment

The following classes of containment, described by Dr. Silverman at an earlier meeting, were reviewed briefly:

Class	Rating	C Value
1. No containment	0	large
2. General building contraction(less than 0.2 psi) with controlled release and provisions for recirculating accumulated fission products through gas cleaners by recycling enclosure gases through collector.	1	intermediate
3. Containment with controlled release for pressures beyond 0.2 to 2 psi with effluent cleaner for all fission products accumulated below these limits through effective cleaning devices.	2	small
4. High integrity pressure shell capable of containing all pressure rises, and fission products for further treatment.	2	small

Effects of missiles on containers must be considered in the assignment of these ratings or C values.

VI. Population Density

It was noted that the reactor owner has no control over the population density in the area. The establishment of a reactor facility may in itself bring about an increase in population density in the surrounding area.

It was agreed that the following statement expresses the Subcommittee's view: "The population distribution should be such that accidental leakage of radioactivity shall not cause the product of the average dose times the

number of people exposed to 1 rem or more to exceed 40 times the reactor power in thermal megawatts". This is for a ten-year period - or 4 roentgen units per megawatt, per year, for ten years.

* * * * * * * * * * * * *

EXCLUSION AREA

There was considerable discussion about the necessary exclusion area around reactors of different powers. Although it is obvious that the selection of certain arbitrary distances for reactors of various powers may be a simple solution, some thought should be given to the basic reason for an exclusion area.

It has been generally stated that exclusion area is for the purpose of protecting against gamma shine in case of accidents and also to give a certain amount of time for warning, evacuation, or other alleviating measures. Mr. Downes made a point that for protection against gamma shine from an unshielded container full of fission products the exclusion area should be approximately three-quarters of a mile. This is for a 500 Mw reactor. He made the point that there is no significant difference in the distance for half versus all of the fission products.

After considerable discussion the Subcommittee generally agreed that the basic principle of an exclusion area should be for the protection of the public outside of it from the gamma shine. The exclusion distance should be such that for the uniform distribution of 100 per cent (or somewhat less) of the gross fission products within the container the dose at this distance would be (according to our notes the Subcommittee did not agree upon any definite number but values of the order of 25 rem

and 100 rem were mentioned). Because the greatest part of the dose is delivered in the first hour the actual time to be specified for the accumulation of the dose is not particularly sensitive, but some number should be arrived at. Values for this should range from 4 to 24 hours.

The Subcommittee was of the opinion that shielding within the container could be substituted for distance even to the extent of reducing the exclusion distance to substantially zero provided that the leakage rate was sufficiently low. The view was expressed that it might be necessary to specify some small exclusion area even with adequate shielding.

The Subcommittee discussed the question of whether or not any activity should be permitted in the exclusion zone without reaching any firm recommendations. One view was that there would be no objection to a large heavy industrial installation, such as an oil refinery, within the exclusion area. It was the consensus that a certain amount of activity could be permitted within this area but that these activities should be limited to those in which persons could be under control and evacuated rapidly in event of accidents.

For comparison purposes the AEC exposure history for a nine-year period was cited.

Number of persons exposed	Exposure range (rem)
187,000	0-1
8,500	1-5
560	5-10
73	10-15
19	> 15

The meeting was adjourned at 4:15 p.m.

SITE CRITERIA

In general the location, design, construction, and operation should be such that all prudent safely conservative principles should be observed to prevent as far as possible injury to persons in the case of an accident. Although the site of a reactor cannot be properly assessed independently of the reactor and its use, nevertheless, it is useful to evaluate these factors independently as far as possible and the guide lines are set out in the following this in view.

The site has certain characteristics independent of the reactor, including population density, meteorology, geology, hydrology, and seismology. Of these the population density is presently the main consideration and in many cases the meteorology is the second most important aspect. The other factors are not to be neglected but are perhaps more easily taken account of in the design. As a guide for the decision that the site, independent of the reactor type, use, and containment, is suitable the number of roentgen units* accumulated by the population surrounding the reactor for an assumed release of gross fission products of .3 per cent during average nighttime meteorological conditions shall be not greater than 40 per megawatt thermal power.

This guide may be modified by appropriate factors which consider the type of the reactor, the experience with the use of this type of reactor, the usage of the reactor, the type of containment used, the amount and

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*Roentgen units are defined as the number of people affected multiplied by their equivalent whole body dose in rem from 1 to 1000. Doses greater than 1000 are counted as 1000, doses less than 1 rem are neglected.

kind of leakage which is specified for the containment and the appropriate geological, hydrological, and seismological factors. Credit also may be claimed for adequate warning and effective use of counter measures. These appropriate factors will be proposed by the applicant and will be accepted or modified by the appropriate hazard evaluation authorities.

Exclusion Area. The exclusion distance will be such that in the event of the uniform dispersion of 100 per cent of the gross fission products in the reactor within the containment structure or building, the dose accumulated by a person at the minimum exclusion distance due to gamma rays in 24 hours will be not more than 100 rem. Appropriate shielding can replace distance.

§ 997.89 Counterparts.

This agreement may be executed in multiple counterparts, and when one counterpart is signed by the Secretary, all such counterparts shall constitute, when taken together, one and the same instrument as if all such signatures were contained in one original.

§ 997.90 Additional parties.

After the effective date of the agreement, any handler may become a party hereto if a counterpart hereof is executed by him and delivered to the Secretary. This agreement shall take effect as to such new contracting party at the time such counterpart is delivered to the Secretary, and the benefits, privileges, and immunities conferred by this agreement shall then be effective as to such new contracting party.*

§ 997.91 Request for order.

Each signatory handler hereto requests the Secretary to issue an order pursuant to the act regulating the handling of filberts grown in the States of Oregon and Washington in the same manner as provided in this agreement.*

Dated: May 19, 1959.

ROY W. LENNARTSON,
Deputy Administrator;
Marketing Services.

[F.R. Doc. 59-4351; Filed, May 22, 1959; 8:46 a.m.]

Agricultural Research Service

[7 CFR Part 301]

EXTENSION OF KHAPRA BEETLE QUARANTINE TO TEXAS

Notice of Public Hearing and of Proposed Rule Making

The Administrator of the Agricultural Research Service has information that the khapra beetle (Trogoderma granarium Everts), a dangerous insect not heretofore widely prevalent or distributed within or throughout the United States, but which previously has been found to exist in certain parts of the States of Arizona, California, and New Mexico, has recently been discovered in certain parts of the State of Texas.

Notice is hereby given that it is proposed under the authority of section 8 of the Plant Quarantine Act of 1912, as amended, and section 106 of the Federal Plant Pest Act (7 U.S.C. 161, 150ee), to quarantine the State of Texas and to prohibit or restrict the movement from Texas into or through any other State, Territory, or District of the United States of (a) all grains and grain products (including, but not limited to, barley, corn, oats, rye, and wheat) whether moved as such or in connection with other articles; (b) dried seeds and seed products of field and vegetable crops (including, but not limited to, alfalfa seed, cottonseed, cottonseed meal and cake, flax seed, sorghum seed, soybean meal, pinto beans, and black-eyed peas); (c) bags and bagging (including, but not limited to, those made of burlap or cotton); (d) dried milk, dried blood, fish meal, and meat scraps; and (e) any other article which by reason of infestation or exposure constitutes a hazard of spreading the khapra beetle; as such articles are defined in regulations supplemental to 7 CFR 301.76.

A public hearing will be held before a representative of the Agricultural Research Service in the Second Floor Auditorium of the Dallas Power and Light Company, 1506 Commerce Street, Dallas, Texas, at 10 a.m., June 23, 1959, at which hearing any interested person may appear and be heard, either in person or by attorney, on the proposals. Any interested person who desires to submit written data, views, or arguments on the proposals may do so by filing the same with the Director of the Plant Pest Control Division, Agricultural Research Service, U.S. Department of Agriculture, Washington 25, D.C., on or before June 23, 1959, or with the presiding officer at the hearing.

Further, notice is hereby given under section 4 of the Administrative Procedure Act (5 U.S.C. 1003) that if it is determined, after hearing, that the State of Texas should be quarantined as proposed, the Administrator of the Agricultural Research Service is considering amending 7 CFR 301.76 by adding the State of Texas to the States designated therein as quarantined

All persons who desire to submit written data, views, or arguments in connection with the proposed quarantine amendment should file the same with the Director of the Plant Pest Control Division, Agricultural Research Service, U.S. Department of Agriculture, Washington 25, D.C., on or before June 23, 1959, or with the presiding officer at the hearing provided for above.

(Sec. 9, 37 Stat. 318, sec. 106, 71 Stat. 33; 7 U.S.C. 162, 150ee. Interprets or applies sec. 8, 37 Stat. 318, as amended; 7 U.S.C. 161; 19 F.R. 74, as amended)

Done at Washington, D.C., this 19th day of May 1959.

M. R. CLARKSON,
Acting Administrator,
Agricultural Research Service.

[F.R. Doc. 59-4371; Filed, May 22, 1959; 8:49 a.m.]

ATOMIC ENERGY COMMISSION

[10 CFR Chapter 1]

POWER AND TEST REACTORS

Notice of Proposed Rule Making

The Commission is considering the formulation of an amendment to its regulations to state site criteria for evaluation of proposed sites for nuclear power and test reactors and is publishing for comment safety factors which might be a basis for the development of site criteria.

In view of the complex nature of the environment, the wide variation in environmental conditions from one location to another and the variations in reactor characteristics and associated protection which can be engineered into a reactor facility, definitive criteria for general application to the siting problems have not been set forth.

All interested persons are invited to submit comments and suggestions on the following site factors and on development of definitive criteria for evaluation of sites for power and test reactors which might be incorporated in the Commission's regulations. All interested persons who desire to submit written comments and suggestions should send them to the U.S. Atomic Energy Commission, Washington 25, D.C., Attention: Division of Licensing and Regulation, within 30 days after publication of this notice in the FEDERAL REGISTER.

Factors considered in site evaluation for power and test reactors—a. *General.* The construction of a proposed power or test reactor facility at a proposed site will be approved if analysis of the site in relation to the hazards associated with the facility gives reasonable assurance that the potential radioactive effluents therefrom, as a result of normal operation or the occurrence of any credible accident, will not create undue hazard to the health and safety of the public.

There are wide possible variations in reactor characteristics and protective aspects of such facilities which affect the characteristics that otherwise might be required of the site. However, the following factors are used by the Commission as guides in the evaluation of sites for power and test reactors. The fact that a particular site may be deemed acceptable for a proposed reactor facility when evaluated in the early phases of the project, does not determine that the reactor will eventually be given operating approval, or indicate what limitations on operation may be imposed. Operating approvals depend on detailed review of design, construction and operating procedures at the final construction stages.

b. *Exclusion distance around power and test reactors.* Each power and test reactor should be surrounded by an exclusion area under the complete control of the licensee. The size of this exclusion area will depend upon many factors including among other things reactor power level, design features and containment, and site characteristics. The power level of the reactor alone does not determine the size of the exclusion area. For any power or test reactor, a minimum radius on the order of one-quarter mile will usually be found necessary. For large power reactors a minimum exclusion radius on the order of one-half to three-quarter miles may be required. Test reactors may require a larger exclusion area than power reactors of the same power.

c. *Population density in surrounding areas.* Power and test reactors should be so located that the population density in surrounding areas, outside the exclusion zone, is small. It is usually desirable that the reactor should be several miles distant from the nearest town or city and for large reactors a distance of 10 to 20 miles from large cities. Where there is a prevailing wind direction it is usually desirable to avoid locating a power or test reactor within several miles upwind from centers of population.

Nearness of the reactor to air fields, arterial highways and factories is discouraged.

d. *Meteorological considerations.* The site meteorology is important in evaluating the degree of vulnerability of surrounding areas to the release of airborne radioactivity to the environment. Capabilities of the atmosphere for diffusion and dispersion of air-borne release are considered in asessing the vulnerability to risk of the area surrounding the site. Thus a high probability of good diffusion conditions and a wind direction pattern away from vulnerable areas during periods of slow diffusion would enhance the suitability of the site. If the site is in a region noted for hurricanes or tornadoes, the design of the facility must include safeguards which would prevent significant radioactivity releases should these events occur.

e. *Seismological considerations.* The earthquake history of the area in which the reactor is to be located is important. The magnitude and frequency of seismic disturbances to be expected determine the specifications which must be met in design and construction of the facility and its protective components. A site should not be located on a fault.

f. *Hydrology and geology.* The hydrology and geology of a site should be favorable for the management of the liquid and solid effluents (including possible leaks from the process equipment). Deposits of relatively impermeable soils over ground water courses are desirable because they offer varying degrees of protection to the ground waters depending on the depth of the soils, their permeability, and their capacities for removing and retaining the noxious components of the effluents. The hydrology of the ground waters is important in assessing the effect that travel time may have on the contaminants which might accidentally reach them to the point of their nearest usage. Site drainage and surface water hydrology is important in determining the vulnerability of surface water courses to radioactive contamination. The characteristics and usage of the water courses indicate the degree of risk involved and determine safety precautions that must be observed at the facility in effluent control and management. The hydrology of the surface water course and its physical, chemical and biological characteristics are important factors in evaluating the degree of risk involved.

g. *Interrelation of factors.* All of the factors described in paragraph b through f of this section are interrelated and dictate in varying degrees the engineered protective devices for the particular nuclear facility under consideration, and the dependence which can be placed on such devices. It is necessary to analyze each of the environmental factors to ascertain the character of protection it might afford for operation of the proposed facility or the kind of restrictions it might impose on the proposed design and operation.

Dated at Germantown, Md., this 19th day of May 1959.

A. R. Luedecke,
General Manager.

[F.R. Doc. 59-4342; Filed, May 22, 1959; 8:45 a.m.]

Minutes

of

ENVIRONMENTAL SUBCOMMITTEE

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

held on

August 23, 1960

at

Washington, D. C.

ATTENDANCE

ACRS:	C. R. McCullough, Chairman
	W. P. Conner
	F. Gifford
	K. R. Osborn
	L. Silverman
	A. Wolman
	R. C. Stratton
	J. J. Fitzgerald, Consultant
	J. B. Graham
	R. F. Fraley
AEC Staff:	C. K. Beck
	J. Lieberman
USWB:	D. Pack

DRAFT 9/20/60 JBG:LSB

Dr. McCullough outlined the scope of the meeting and asked Dr. Beck to summarize his revised criteria for site selection (draft #4, 8/5/60) including some statement of the underlying philosophy.

In his early remarks Dr. Beck stressed that his draft criteria did not explicitly deal with the probabilities of accidents but it is implied that these probabilities are very low. He then defined the following terms which, for convenience, are set forth below:

Exclusion Area -- An area whose radius is not less than the distance at which total radiation doses received by an individual fully exposed for two hours to the radioactive consequences of the maximum credible accident would be above 25 R (or equivalent). The area should be under the full control of the applicant. Residents subject to ready evacuation are allowed.

Evacuation Area -- An area whose radius is not less than the distance at which total radiation doses received by an individual fully exposed for the entire maximum credible accident would be above 25 R (or equivalent). Total population not to exceed 10,000 people and no more than 2,000 in any 45° sector.

City Distance -- Distance from reactor to nearest fringe of high density population of a substantial city (above 10,000) which must not be less than distance at which total radiation doses received by a person exposed for the entire maximum credible

accident would be above 10 R or equivalent. The real basis, however, for this criterion is an uncontained "puff" release resulting in a LD-50 dose at the city boundary.

Dr. Beck observed that the meteorological parameters used in his tabulation of existing and proposed reactors were reviewed with USWB. Don Pack pointed out that this was so but there had been some slippage in the values given for U.S. Average conditions (see Item 6, Appendix A).

The criteria presented are intended to be applied to pressurized water and boiling water power reactors. Mr. Osborn thought that this might be unfair to the pressurized water reactors. Dr. Beck believes that the probabilities of major accidents in either type is sufficiently low that one can neglect the differences in the numbers themselves.

Dr. Conner asked if Dr. Beck's criteria considered genetic damage and the reply was that it was considered but at large distances (low doses) all reactors look the same. Also, the Sutton equation is not valid for these large distances. Dr. Conner expressed the view that the life shortening phenomena should also be considered.

Dr. Gifford observed that there are two bad things the criteria might do: the first would be to allow a reactor to be sited too close to population, and the second would be to exclude reactors which should be allowed to locate at a lesser distance from population.

Dr. McCullough reviewed the concept of equivalent dose on the basis of radiation induced leukemias.

20 leukemias due to 10^{+6} man rem (whole body)
5 " " " " " " (thyroid)
10 " " " " " " (bone dose)

In executive session it was agreed that the ACRS should take a strong position to the effect that quantitative criteria cannot be written at this time. There was sentiment in favor of describing the philosophy which has been used by the Committee in arriving at safety judgments.

Reference 4: Reactor Site Criteria Report to General Manager by Director, Division of Licensing & Regulations, Document Date May 25, 1959, ML021960199

ATOMIC ENERGY COMMISSION

REACTOR SITE CRITERIA

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

THE PROBLEM

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

SUMMARY

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

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

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

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

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

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

8. Proposed criteria (Appendix "D") have been prepared that describe the bases upon which the suitability of proposed reactor sites can be judged. As a beginning point, the criteria define three bench marks, stated in terms of areas and distances, for evaluation of proposed sites for a reactor of any given power level. These are **(1)** an exclusion area over which the licensee controls the access; (2) a zone surrounding the exclusion area in which the density of population is sufficiently low to permit evacuation in case of a catastrophic accident; and (3) a distance to the nearest population center in which more than 25,000 people reside. These areas and distances are determined upon the following assumptions: **(1)** in establishing the exclusion and evacuation distances, the amount of radioactivity released to the environment will not exceed that expected from the accident considered to be "the maximum credible accident"; (2) within the exclusion area the operator will have full control and may take whatever steps are necessary to protect any people who may be therein; (3) the radiation dose to persons within

the evacuation area may be limited by evacuation or other counter measures sufficiently to prevent immediate or early manifestation of radiation injury; and 4) the population center distance is calculated on the assumption that persons in nearby centers of population would not be lethally exposed in the event of an accident similar to the maximum credible accident but in which no containment or retention whatever of the released fission products were accomplished by the reactor building. Iodine doses such as those specified (in later sections) on the basis of these premises, if actually received by people, do not preclude the possibility of the production of a number of cases of leukemia or cancer in later years. However, it is believed that in view of the small probability of occurrence of accidents comparable to the "maximum credible accident", the hazard from such effects as well as from genetic effects is reasonably small. The criteria then provide for adjustment of these bench mark distances in each case in accordance with the unique features and circumstances of that individual reactor project. The proposed rule makes it clear that the bench mark distances are only a beginning point for preliminary guidance and have to be considered along with other equally important factors.

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

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

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

the criteria could be revised to reflect either more or less conservatism with respect to degree of isolation to be required in future reactor projects.

STAFF JUDGMENTS

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

RECOMMENDATION

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

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

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

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

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

e. Note that this paper is unclassified.

Reference 4: Reactor Site Criteria Report to General Manager by Director, Division of Licensing & Regulations, Document Date May 25, 1959, ML021960199

LIST OF ENCLOSURES

APPENDIX "A" -Background

APPENDIX "B"
Conservatisms in the Assumptions and Factors Used in Calculating the Consequences of the Maximum Credible Accident

APPENDIX "C-l" -Letter dated Sept. 26, 1960 from Leslie Silverman to John McCone

APPENDIX "C-2"-Letter dated 10/22/60 from Leslie Silverman to John McCone

APPENDIX "D" -10 CFR Part 51 -Reactor Site Criteria

ANNEX "1" to APPENDIX "D" APPENDIX "A" -
Calculation of Bench Mark Areas and Distances

ANNEX "2" to APPENDIX "D" APPENDIX "A" (alternate 1) -Calculation of Bench Mark Areas and Distances (concentration limits)

ANNEX "3" to APPENDIX "D" APPENDIX "A" (alternate 2) -Table of Bench Mark Areas and Distances

APPENDIX "E" -Bench Marks for Selected Reactors

APPENDIX "A"

BACKGROUND

Introduction

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

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

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

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

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

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

4. It is generally recognized that uncontrolled release to the atmosphere of the radioactive contents of a reactor system located in a densely populated area would result in public disaster. This awareness has led to the provision in the past of a considerable isolation area surrounding reactor installation. This was done on the theory that if enough distance was provided between

a reactor and the perimeter of the controlled area, little or no Jeopardy to the public would be involved.

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

6. Any further reduction in the concepts of isolation and containment for reactors will be largely dependent upon the ability to assess with more certainty the circumstances and conditions under which loss of control of radioactive inventory might arise and the possible consequences of such an accident. The process of hazard analysis and site selection at this stage of technology is not a precise science for the many variables involved are not precisely known nor has experience been sufficient to provide exact knowledge about the degree of conservatism that exists in past assumptions and guiding design criteria.

Present Practices in Site Evaluation

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

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

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

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

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

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

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

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

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

10. All the factors described are interrelated and dictate in varying degrees the engineered protective safeguards required for an individual facility. Therefore, site evaluation also includes consideration of the general features of the reactor plant including power level, general plan of utilization and the safeguards planned to preclude or minimize inadvertent release of radioactive effluents.

11. An analytical test of the safeguards provided by site location and plant design is made through evaluation of a postulated accident, having consequences not expected to be exceeded by any other accident arising out of any other credible circumstances. Analysis is

made of the consequences in terms of possible radiation exposure both to personnel at the facility and to the inhabitants of the surrounding public area. The conservatism of the assumptions made in arriving at the results and the acceptability of characteristics attributed to the safeguards provided are considered in assessing the numerical values derived. The judgment made is thus highly subjective. The many variables involved are not precisely known nor has experience been sufficient to provide exact knowledge about the degree of conservatism that exists in past design assumptions and guiding criteria.

History of the Problem

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

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

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

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

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

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

13. Notwithstanding these deterrents to the formulation of definitive site criteria the AEC has been attempting to establish a more objective approach to site evaluation. For example, the AEC issued for public comment and published in the Federal Register on May 23, 1959, a notice of proposed rule-making that set forth general criteria for evaluation of sites for power and test reactors. That notice resulted in widespread reactions from the industry with definite indication of opposition to formal siting regulations. AEC-R 2/20 contains excerpts of comments which the AEC received in writing together with comments made at meetings of the Technical Appraisal Task Force on Nuclear Power of the Edison Electrical Institute (EEI) on June **1,** 1959, and the Atomic Industrial Forum on June 30, 1959.

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

15. A draft of criteria along the lines of the proposed regulation was submitted to the ACBS for review and comments. A copy of that earlier draft is being circulated as AEC-R 2/22. The ACRS by letter to the Chairman, AEC, dated September 26, 2960 (Appendix "C-l") expressed the view that the proposed criteria could be developed into a useful contribution to nuclear safety studies but the criteria document should not be given the status of a Commission regulation. A similar recommendation, together with additional comments, was made by the ACRS in a letter of October 22, 1960 to Chairman McCone. (Appendix "C-2")

Reference 4: Reactor Site Criteria Report to General Manager by Director, Division of Licensing & Regulations, Document Date May 25, 1959, ML021960199

DISCUSSION

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

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

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

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

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

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

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

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

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

b. Evacuation distance - The greatest distance from the facility at which the total radiation dose received by an individual located at such distance and exposed during the whole course of the

maximum credible accident to the radioactive cloud resulting from the accident would be 25 rem to the whole body or 300 rem to the thyroid from radioactive-iodine exposure.

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

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

20. Provisions are made in the criteria for consideration of other relevant factors as well as the bench mark distances. The application of these criteria depends to a substantial degree on the subjective evaluative judgments of the person responsible for final approval of a reactor site. Thus adoption of these criteria will not provide fully objective procedures for site selection. Rather these procedures define bench mark distances as a beginning point in the evaluation process. This would be in contrast to the methods which have been utilized to the present time. There has been no common point of departure and hence this entire process has depended upon subjective judgment.

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

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

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

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

22. The disadvantages to this form of presentation are:

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

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

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

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

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

24. The disadvantages to this form of presentation are:

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

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

25. A third method of presenting the proposed criteria is shown by Annex 3 to Appendix **"D".** In this annex, the bench mark distance factors as a function of power level have been calculated

and presented in the form of a table. The basis upon which the table has been computed has been omitted. The advantage of such a scheme is its simplicity. A principal disadvantage is that the fundamental bases for establishing the bench marks are hidden. Of course, those bases could be explained by press releases, speeches, etc., but the staff feels that the best place to explain them is in the regulation itself.

26. After consideration of the relative merits of the various ways in which the criteria might be expressed, it is the opinion of the staff of the Division of Licensing and Regulation that the bench mark calculations as presented in the form shown in Annex 1 to Appendix **"D"** (combined with a precalculated table) wherein the distance factors are defined in terms of reference dose limits, will best serve the interests of both the nuclear industry and the public and most clearly defines the basis upon which the AEC intends to evaluate proposed reactor locations.

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

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

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

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

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

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

APPENDIX "B"

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

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

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

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

3. The conservatism reflected in the reactor plants is revealed through the analytical technique of postulating a severe accident condition and then evaluating the ability of the plant to remain under control and, through the safeguards provided, including location, prevent or minimize the effects of release of hazardous radioactive effluents. Whereas the exact probability of a major release cannot be predicted, it is possible to arrive at a judgment on site suitability through analysis of the conservatism reflected both in design and the assumptions made in calculating the consequences of a major accident. This in brief is the general approach that has been used by the AEC and the ACRS to arrive at their judgments on applications for construction permits.

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

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

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

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

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

(1) Reactivity control mechanisms fail to function, and

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

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

These prior failures need not occur for the case of a 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 should occur and coolant is lost, fuel melting requires that:

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

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

9. Despite such safeguards as those described above, if a major release of fission products to the environment should occur, estimations of the exposure doses which might result to persons

offsite are extremely difficult to make because of the complex and interwoven technical effects involved. Although the amount of each kind of radioactive material present in a reactor system can be estimated fairly closely, as a function of the power level history, how much of this material would be released as a result of an accident is highly unpredictable. Quantities in the order of 10 -30% of the total inventory have been assumed in the past. Experimental data would indicate these values to be conservative but the exact release can vary so much from reactor system to system and with the detailed nature of an accident that the exact degree of conservatism is not known. Further, there is a multiplicity of possible patterns of atmospheric dispersal whereby these radioactive materials can be transported to areas beyond the site boundary and those patterns can vary markedly from one reactor location to another.

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

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

12. It is from this complexity of interwoven technical parameters that criteria for use in the selection of sites has been formulated. While these criteria represent a considerable simplification of the many complex phenomena involved, they represent the same very conservative approach to site selection that has characterized such evaluations in the past. The

fundamental assumptions upon which the proposed bench mark distances are based with estimates of the degree of conservatism represented in each case are as follows:

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

Reference 4: Reactor Site Criteria Report to General Manager by Director, Division of Licensing & Regulations, Document Date May 25, 1959, ML021960199

APPENDIX "C-I"

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

September 26, 1960

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

Subject: CRITERIA FOR EVALUATION IOF REACTOR SITES

Dear Mr. McCone:

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

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

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

Sincerely yours,

Leslie Silverman
Chairman

Reference 4: Reactor Site Criteria Report to General Manager by Director, Division of Licensing & Regulations, Document Date May 25, 1959, ML021960199

APPENDIX “C-2”

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
UNITED STATES ATOMIC ENERGY COMMISSION
WASHINGTON 25. D. C.
October 22, 1960

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

Subject: REACTOR SITE CRITEREA

Dear Mr. McCone:

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

The Committee believes that the officially endorsed numbers could stifle progress toward a better selection of numbers. The ideas and interpretations from applicants themselves have played a major part in the formulation of the current bases for site evaluation. It would be a significant loss to stop the flow of new ideas from the applicants. The Committee also believes that it is possible that the appearance of quantitative numbers in a Federal regulation or policy statement will reduce the continual awareness of the applicant that he has assumed a responsibility to be alert to and to act on unforeseen disadvantages of a site even after the site has been approved. The Committee, therefore, advises that a quantitative statement of site criteria not be included in Federal regulations.

These comments do not mean that the ACRS has no bases for judging the adequacy of sites. They merely emphasize that site selection is still largely a matter of judgment. Inasmuch as the ACRS has been making site and reactor evaluations, it may be helpful to review the framework on which these judgments are being made. It is a prerequisite, of course, that the reactor be

carefully and competently designed, constructed, and operated. It should be inspected during all these stages in a manner to assure preservation of the intended protection of the public. Also, these factors are applicable only to those reactors on which experience has been developed. Reactors which are novel in design, unproven as prototypes, or which do not have adequate theoretical and experimental or pilot plant experience belong at isolated sites - the degree of isolation required depending on the amount of experience which exists.

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

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

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

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

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

Incidentally, the concept has been proposed by others that the damage to people from reactor accidents can be accepted if it is no greater than that experienced in other industries. We reject this suggestion as premature, and follow rather the concept, that the consequences of reactor accidents must be less than this. The reasons for this rejection are twofold: First, we do not have sufficient information on the probability of reactor accidents to make use of this concept in site evaluations. We do use, of course, the fact that the probability of a serious accident is very low. Second, we recognize that the atomic power business has not yet reached the status of supplying an economic need in a manner similar to that of more mature industries; and, therefore, arguments for taking conventional risks for the greater good of the public are somewhat weak. At-the same time, we do not want to imply that the restrictions placed on site locations during the developmental period of atomic power will necessarily be carried over to the period of maturity of the atomic power industry.

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

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

Source Term

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

Dispersal of the Radioactivity

1) Meteorology

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

2) Hydrology

Reference 4: Reactor Site Criteria Report to General Manager by Director, Division of Licensing & Regulations, Document Date May 25, 1959, ML021960199

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

Effect of Radioactivity on People

The upper limit to the exposure to a member of the public in the generalized accident should be no higher than the maximum once-in a-lifetime emergency dose. Such a level has not been established by AEC. We are arbitrarily using a figure of about 25 r whole body or equivalent integrated dose for this level. This figure is mentioned in Handbook 59 of the National Bureau of Standards, pages 69-70. Since the iodine dose is often controlling, we are tentatively considering a thyroid dose limitation of 200 - 300 rads. The dosage so far mentioned refers to limits to people when the people are considered as independent individuals. We believe that it is essential that the Atomic Energy Commission attempt to confirm through its staff or its advisors in this field that this suggested value of 25 r whole body or equivalent is without significant biological effect on the individuals who might be subjected to this dose from the generalized accident.

When large numbers of individuals are exposed to radiation, another limit also exists because of genetic effects and because of the statistical nature of induced leukemia and the shortening of the life span. The limits of exposure to large groups of people are better expressed in terms of integrated man-rems. We are considering using a figure of 4 x 106 man-rems for this limit for the people who might be exposed to radiation doses falling between 1 and 25 rems. This figure of 4 x 106 man-rems is roughly equal to the dose received from natural background by a million people during their reproductive lifetime.

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

The Committee wishes to emphasize again that the numbers which have been used in discussion of the generalized accident should not be formalized into regulations or Commission policy. The Committee wishes to acknowledge the help it has received from the Hazards Evaluation Branch in this matter and suggests that these individuals be encouraged to present

as technical papers, but not as regulations, a complete description of their working approach to making judgments on the adequacy of proposed reactor sites. Such a paper, of course, would have the status of the opinion of an informed technical individual, but would not imply Committee approval, nor would it have the rigidity of a Commission policy statement.

Sincerely yours,

Leslie Silverman
Chairman

Reference 4: Reactor Site Criteria Report to General Manager by Director, Division of Licensing & Regulations, Document Date May 25, 1959, ML021960199

Appendix "D"

ATOMIC ENSERGY COMMISSION
10 CFR Part 51
REACTOR SITE CRITERIA

Notice of Proposed Rule Making

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

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

Recognizing that it is not possible at the present time to define site criteria with sufficient definiteness to eliminate the exercise of agency judgment, the proposed rule set forth below is designed primarily to identify a number of factors considered by the Commission and the general criteria which are utilized as guides in evaluating proposed sites. Through the use of certain assumptions and general calculational techniques set forth in Appendix "A", the proposed rule also attempts to establish a common starting point from which location factors can be assessed by the Commission, the applicant and other interested parties.

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

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

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

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

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

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

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

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

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

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

Reference 4: Reactor Site Criteria Report to General Manager by Director, Division of Licensing & Regulations, Document Date May 25, 1959, ML021960199

As previously indicated, these bench marks are only a starting point in the evaluation of a proposed reactor location. The proposed rule specifies that the commission will also consider physical characteristics of the site, such as seismology, meteorology, hydrology, and geology; and characteristics of the reactor, such as maximum power level, proposed use, engineering safeguards, and unique design features. The over-all judgment is based on these features as well as the population density factors represented by the bench marks. Obviously, as specifically indicated in the proposed rule, the Commission may approve a proposed site which does not meet the bench marks or may disapprove a proposed site which does meet the bench marks.

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

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

(List of Section Headings)

AUTHORITY:

GENERAL PROVISIONS

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

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

§ 51.3 Definitions. As used in this part:

(a) "Exclusion area" means the area surrounding the reactor, access to which is under the full control of the reactor owner. This area may be traversed by a highway or railroad, provided such highway or railroad is not so close to the facility as to interfere with normal operations, and provided appropriate and effective arrangements are made to control traffic on the highway or railroad to protect the public health and safety. Residence within the exclusion area shall be

minimal and residents shall be subject to ready removal in case of necessity to minimize hazard. Activities unrelated to operation of the reactor may be permitted in an exclusion area provided that no significant hazards to the public health and safety will result from the location of the activity in the exclusion area.

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

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

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

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

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

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

SITE EVALUATION FACTORS

§ 51.10 Factors to be Considered When Evaluating Sites. In determining the acceptability of a site for a power or testing reactor, the Commission will take the following factors into consideration:
(a) Population density and use characteristics of the site and its environs, including, among other things, the exclusion area, evacuation area and population center distance.

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

Reference 4: Reactor Site Criteria Report to General Manager by Director, Division of Licensing & Regulations, Document Date May 25, 1959, ML021960199

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

§ 51.11 Application of Site Evaluation Factors. The method by which the Commission will evaluate the factors described in 9 51.10 is as follows:

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

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

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

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

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

For example:

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

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

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

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

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

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

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

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

Reference 4: Reactor Site Criteria Report to General Manager by Director, Division of Licensing & Regulations, Document Date May 25, 1959, ML021960199

ANNEX 1 TO APPENDIX "D" .
APPENDIX A

Calculation of Bench Mark Areas/and Distances

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

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

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

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

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

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

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

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

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

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

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

u is the wind velocity

n is the atmospheric stability parameter

C_y and C_z are horizontal and vertical diffusion parameters respectively

π is a constant 3.1416.

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

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

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

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

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

Reference 4: Reactor Site Criteria Report to General Manager by Director, Division of Licensing & Regulations, Document Date May 25, 1959, ML021960199

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

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

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

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

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

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

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

r is the distance in meters

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

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

t is the exposure time in seconds.

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

Annex 2 to Appendix "D
Appendix A (alternate 1)
Calculation of Bench Mark Areas and Distances (concentration limits)

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

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

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

where:

d is the distance

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

u is the wind velocity

n is the atmospheric stability parameter

C_y and C_z, are horizontal and vertical diffusion parameters

π is the constant 3.1416

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

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

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

Reference 4: Reactor Site Criteria Report to General Manager by Director, Division of Licensing & Regulations, Document Date May 25, 1959, ML021960199

Annex 3 to Appendix "D"
Appendix A (Alternate 2)
Table of Bench Mark Areas and Distances

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

TABLE OF BENCH MARK LOCATION DISTANCES

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

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

APPENDIX "E"
BENCH MARKS FOR SELECTED REACTORS

		Exclusion Area		Evacuation Area		Population Center Distance	
(MWt)	Reactor	Bench Mark Distance (miles)	Actual Distance (miles)	Bench Mark Distance (miles)	Actual Pop. Density in Bench Mark Area (people/sq.mi.)	Bench Mark Distance (miles)	Actual Distance (miles)
630	Dresden	.33	.5	7.4	38	9.9	14
585	Con. Ed.	.31	.3	7.0	403	9.4	17
485	Yankee	.28	.5	6.2	33	8.3	21
300	PRDC	.23	.75	4.5	24	6.1	7.5
270	PWR	.23	.4	4.2	298	5.7	7.5
240	Consumers	.22	.5	3.9	28	5.2	135.0
240	Hallam	.22	.25	3.9	10	5.2	17
203	Pathfinder	.21	.5	3.5	25	4.6	3.5
202	PG&E	.21	.25	3.5	172	4.6	3
200	ICBWR	.21	.2	3.5	86	4.6	10
115	Phila. Elec.	.19	.57	2.4	29	3.2	21.0
60	NASA	.16	.57	1.6	53	2.1	3
60	CVTR	.16	.5	1.6	12	2.1	25
60	Jamestown (Orig. site)	.16	.3	1.6	1200	2.1	0.5
60	Jamestown (New site)	.16	.3	1.6	66	2.1	2.4
58	ElK River	.16	.23	1.5	40	2.0	20.0
50	VBWR	.15	.4	1.4	23	1.9	15.0
48	Piqua	.15	.14	1.4	960	1.8	27.0
40	Pt. Loma	.14	.25	1.2	0	1.6	3

Welfare by the Federal Food, Drug, and Cosmetic Act (sec. 408 (b), (e), 68 Stat. 511, 514; 21 U.S.C. 346a (b), (e)) and delegated to the Commissioner of Food and Drugs by the Secretary (25 F.R. 8625) it is proposed by the Commissioner, on his own initiative, that the regulations for tolerances for pesticide chemicals in or on raw agricultural commodities (21 CFR 120.6) be amended by inserting in alphabetical order, the item, "copper sulfate pentahydrate" to the list of exempted copper compounds in paragraph (b)(1). As amended, § 120.6(b)(1) would read as follows:

§ 120.6 Exemptions from the requirement of a tolerance.

* * * * * *

(b) * * *

(1) The following copper compounds: Bordeaux mixture, copper acetate, basic copper carbonate (malachite), copperline mixtures, copper oleate, copper oxychloride, copper silicate, copper sulfate basic, copper sulfate monohydrate, copper sulfate pentahydrate, copper-zinc chromate, cuprous oxide, tetra copper calcium oxychloride.

A person who has registered or who has submitted an application for the registration of an economic poison under the Federal Insecticide, Fungicide and Rodenticide Act containing copper sulfate pentahydrate may request, within 30 days from publication of this proposal in the FEDERAL REGISTER, that the proposal be referred to an advisory committee in accordance with section 408(e) of the Federal Food, Drug, and Cosmetic Act.

Any interested person is invited at any time prior to the thirtieth day from the date of publication of this notice in the FEDERAL REGISTER to file with the Hearing Clerk, Department of Health, Education, and Welfare, Room 5640, 330 Independence Avenue SW., Washington 25, D.C., written comments on the proposal. Comments may be accompanied by a memorandum or brief in support thereof.

All documents shall be filed in quintuplicate.

Dated: February 6, 1961.

[SEAL] JOHN L. HARVEY,
Deputy Commissioner, of Food and Drugs.

[F.R. Doc. 61-1218; Filed, Feb. 10, 1961; 8:49 a.m.]

[21 CFR Part 121]

FOOD ADDITIVES

Notice of Filing of Petition

Pursuant to the provisions of the Federal Food, Drug, and Cosmetic Act (sec. 409(b)(5), 72 Stat. 1786; 21 U.S.C. 348(b)(5)), notice is given that a petition has been filed by E. I. du Pont de Nemours and Company, 1007 Market Street, Wilmington 98, Delaware, proposing the issuance of a regulation to provide for the safe use of dibutyl phthalate, dicyclohexyl phthalate, and toluenesulfonamide formarlehyde resin in nitrocellulose lacquers used on cellulosic substrates to produce heat-sealing packaging materials for various types of solid foods.

Dated: February 7, 1961.

[SEAL] J. K. KIRK,
Assistant to the Commissioner of Food and Drugs.

[F.R. Doc. 61-1230; Filed, Feb. 10, 1961; 8:50 a.m.]

[21 CFR Part 121]

FOOD ADDITIVES

Notice of Filing of Petition

Pursuant to the provisions of the Federal Food, Drug, and Cosmetic Act (sec. 409(b)(5), 72 Stat. 1786; 21 U.S.C. 348(b)(5)), notice is given that a petition has been filed by Harry Miller Corporation, Fourth and Bristol Streets, Philadelphia 40, Pennsylvania, proposing the issuance of a regulation to provide for the safe use of mineral oil as a component of lubricants used in the drawing and forming of sanitary can components for food packaging.

Dated: February 7, 1961.

[SEAL] J. K. KIRK,
Assistant to the Commissioner of Food and Drugs.

[F.R. Doc. 61-1231; Filed, Feb. 10, 1961; 8:50 a.m.]

[21 CFR Part 121]

FOOD ADDITIVES

Notice of Filing of Petition

Pursuant to the provisions of the Federal Food, Drug, and Cosmetic Act (sec. 409(b)(5), 72 Stat. 1786; 21 U.S.C. 348(b)(5)), notice is given that a petition has been filed by Sterwin Chemicals, Inc., a subsidiary of Sterling Drug, Inc., 1450 Broadway, New York 18, New York, proposing the issuance of a regulation to provide for the safe use of a combination of bithionol and methiotriazamine as a coccidiostat in chicken feeds.

Dated: February 7, 1961.

[SEAL] J. K. KIRK,
Assistant to the Commissioner of Food and Drugs.

[F.R. Doc. 61-1232; Filed, Feb. 10, 1961; 8:50 a.m.]

ATOMIC ENERGY COMMISSION

[10 CFR Part 100]

REACTOR SITE CRITERIA

Notice of Proposed Guides

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

Judgment of suitability of a reactor site for a nuclear plant is a complex task. In addition to normal factors considered for any industrial activity, the possibility of release of radioactive effluents requires that special attention be paid to physical characteristics of the site, which may cause an incident or be of significant importance in increasing or decreasing the hazard resulting from an incident. Moreover, the inherent characteristics and the specifically designed safeguard features of the reactor are of paramount importance in reducing the possibility and consequences of accidents which might result in the release of radioactive materials. All of these features of the reactor plus its purpose and method of operation must be considered in determining whether location of a proposed reactor at any specific site would create an undue hazard to the health and safety of the public.

Recognizing that it is not possible at the present time to define site criteria with sufficient definiteness to eliminate the exercise of agency judgment, the proposed guides set forth below are designed primarily to identify a number of factors considered by the Commission and the general criteria which are utilized as guides in evaluating proposed sites.

The basic objectives which it is believed can be achieved under the criteria set forth in the proposed guides, are:

(a) Serious injury to individuals offsite should be avoided if an unlikely, but still credible, accident should occur.

(b) Even if a more serious accident (not normally considered credible) should occur, the number of people killed should not be catastrophic.

(c) The exposure of large numbers of people in terms of total population dose should be low. The Commission intends to give further study to this problem in an effort to develop more specific guides on this subject. Meanwhile, in order to give recognition to this concept the population center distances to very large cities may have to be greater than those suggested by these guides.

Notice is hereby given that adoption of the following guides is contemplated. All interested persons who desire to submit written comments and suggestions for consideration in connection with the proposed guides should send them to the Secretary, United States Atomic Energy Commission, Washington 25, D.C., Attention: Director, Division of Licensing and Regulation, within 120 days after publication of this notice in the FEDERAL REGISTER.

GENERAL PROVISIONS

SITE EVALUATION FACTORS

GENERAL PROVISIONS

§ 100.1 Purpose.

It is the purpose of this part to describe the criteria which guide the Commission in its evaluation of the suitability of proposed sites for power and testing reactors subject to Part 50 of this chapter. Because it is not possible to define such criteria with sufficient definiteness to eliminate the exercise of agency judgment in the evaluation of these sites, this part is intended primarily to identify a number of factors considered by the Commission and the general criteria which are utilized as guides in approving or disapproving proposed sites.

§ 100.2 Scope.

(a) This part applies to applications filed under Part 50 of this chapter for construction permits and operating licenses for 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 with additional conservatism to other reactors. For reactors which are novel in design, unproven as prototypes, and do not have adequate theoretical and experimental or pilot plant experience, these criteria will need to be applied more conservatively. This conservatism will result in more isolated sites—the degree of isolation required depending upon the lack of certainty as to the safe behavior of the reactor. It is essential, of course, that the reactor be carefully and competently designed, constructed, operated, and inspected.

§ 100.3 Definitions.

As used in this part:

(a) "Exclusion area" means the area surrounding the reactor, access to which is under the full control of the reactor licensee. 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, 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. Residence 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 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 §§ 50.21(b) or 50.22 of this chapter designed to produce electrical or heat energy.

(e) "Testing reactor" means a "testing facility" as defined in § 50.2 of this chapter.

SITE EVALUATION FACTORS

§ 100.10 Factors to be considered when evaluating sites.

In determining the acceptability of a site for a power or testing reactor, the Commission will take the following factors into consideration:

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

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

(1) The design for the facility should conform to accepted building codes or standards for areas having equivalent earthquake histories. No facility should be located closer than ¼ to ½ mile from the surface location of a known active earthquake fault.

(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. Unless special precautions are taken, reactors should not be located at sites where radioactive liquid effluents might flow readily into nearby streams or rivers or might find ready access to underground water tables.

Where some 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.

(c) Characteristics of the proposed reactor, including proposed maximum power level, use of the facility, the extent to which the design of the facility incorporates well proven engineering standards, and the extent to which the reactor incorporates unique or unusual features having a significant bearing on the probability or consequences of accidental releases of radioactive material.

§ 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 release from the core as illustrated in Appendix "A" of this part, the expected demonstrable leak rate from the containment, and meteorological conditions pertinent to his site to derive an exclusion area, a low population zone and a population center distance. For the purpose of this analysis, the applicant should determine the following:

(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 1⅓ 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 ncessasry because of total integrated population dose considerations.

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

(b)(1) Appendix "A" of this part contains an example of a calculation for hypothetical reactors which can be used as an initial estimate of the exclusion area, the low population zone, and the population center distance.

(2) The calculations described in Appendix "A" of this part are a means of obtaining preliminary guidance. They may be used as a point of departure for consideration of particular site requirements which may result from evaluations of the particular characteristics of the reactor, its purpose, method of operation, and site involved. The numerical values stated for the variables listed in Appendix "A" of this part represent approximations that presently appear reasonable, but these numbers may need to be revised as further experience and technical information develops.

Dated at Germantown, Maryland, this 8th day of February 1961.

For the Atomic Energy Commission.

WOODFORD B. MCCOOL,
Secretary.

APPENDIX "A"

Example of a calculation of reactor siting distances:

1. The calculations of this Appendix are based upon the following assumptions:

a. The fission product release to the atmosphere of the reactor building is 100 percent of the noble gases, 50 percent of the halogens and 1 percent of the solids in the fission product inventory. This release is equal to 15.8 percent of the total radioactivity of the fission product inventory. Of the 50 percent of the halogens released, one-half is assumed to adsorb onto internal surfaces of the reactor building or adhere to internal components.

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

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

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

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

$$X=\frac{Q}{\pi u\sigma_y\sigma_z}$$

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

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

u is the wind velocity

σ_y and σ_z are horizontal and vertical diffusion parameters resp.

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

$u=1$m/sec;
$\sigma_y=[\frac{1}{2}C_y{}^2d^{2-n}]^{1/2}$;
$\sigma_z=[\frac{1}{2}C_z{}^2d^{2-n}]^{1/2}$;
$C_y=0.40$;
$C_z=0.07$;
$n=0.5$

g. The isotopes of iodine were assumed to be controlling for the low population zone distance and population center distance. 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.

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

Isotope	Exclusion Q (curies/megawatt)	Low population Q (curies/megawatt)
I^{131}	0.55	76.4
I^{132}	.68	1.40
I^{133}	1.19	18.5
I^{134}	.72	.91
I^{135}	1.04	5.4

These source terms combine the effects of fission yield under equilibrium conditions, radioactive decay in the reactor building, and the release rate from the reactor building, all integrated throughout the exposure time considered.

i. For the exclusion distance, doses from both direct gamma radiation 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.

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

Isotope	Dose (rem)
I^{131}	329
I^{132}	12.4
I^{133}	92.3
I^{134}	5.66
I^{135}	25.3

k. The whole body doses at the exclusion and low population zone distances due to direct gamma radiation from the fission products released into the reactor building were derived from the following relationships:

$$D=483\frac{Be^{-\mu r}}{4\pi r^2}\int t^{-0.2}dt$$

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

r is the distance in meters

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

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

t is the exposure time in seconds.

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

2. On the basis of calculation methods and values of parameters described above, initial estimates of distances for reactors of various power levels have been developed and are listed below.

Power level (thermal megawatt)	Exclusion distance (miles)	Low population zone distance (miles)	Population center distance (miles)
1500	0.70	13.3	17.7
1200	.60	11.5	15.3
1000	.53	10	13.3
900	.50	9.4	12.5
800	.46	8.6	11.5
700	.42	8.	10.7
600	.38	7.2	9.6
500	.33	6.3	8.4
400	.29	5.4	7.2
300	.24	4.5	6
200	.21	3.4	4.5
100	.18	2.2	2.9
50	.15	1.4	1.9
10	.08	.5	.7

[F.R. Doc. 61-1233; Filed, Feb. 10, 1961; 8:50 a.m.]

FEDERAL AVIATION AGENCY

[14 CFR Parts 600, 601]

[Airspace Docket No. 60-NY-150]

FEDERAL AIRWAYS, CONTROL AREAS AND REPORTING POINTS

Revocation

Pursuant to the authority delegated to me by the Administrator (14 CFR 409.13), notice is hereby given that the Federal Aviation Agency is considering an amendment to Parts 600 and 601 of the regulations of the Administrator, the substance of which is stated below.

Blue Federal airway No. 40 extends from Lebanon, N.H., to Burlington, Vt. The Federal Aviation Agency is considering revoking this airway. It is the policy of this Agency to revoke L/MF airways wherever adequate VOR airways are available, and it appears that the route from Lebanon to Burlington is adequately served by VOR Federal airway No. 151. In addition, the Federal Aviation Agency IFR peak-day airway traffic survey for the period July 1, 1959, through June 30, 1960, shows a maximum of five aircraft movements on Blue 40. Therefore, it appears that the retention of this airway is unjustified as an assignment of airspace. Accordingly, the Federal Aviation Agency proposes to revoke Blue 40 and its associated control areas. Adoption of this proposal would not necessarily result in discontinuance of the low frequency navigational aids associated with Blue 40. Any proposal to discontinue one or more of these aids would be processed in accordance with current Agency procedures. In addition, § 601.4640 relating to reporting points on Blue 40 would be revoked.

Interested persons may submit such written data, views or arguments as they may desire. Communications should be submitted in triplicate to the Chief, Air Traffic Management Division, Federal Aviation Agency, Federal Building, New York International Airport, Jamaica 30, N.Y. All communications received within forty-five days after publication of this notice in the FEDERAL REGISTER will be considered before action is taken on the proposed amendment. No public hearing is contemplated at this time, but arrangements for informal conferences with Federal Aviation Agency officials may be made by contacting the Regional Air Traffic Management Division Chief, or the Chief, Airspace Utilization Division, Federal Aviation Agency, Washington 25, D.C. Any data, views or arguments presented during such conferences must also be submitted in writing in accordance with this notice in order to become part of the record for consideration. The proposal contained in this notice may be changed in the light of comments received.

The official Docket will be available for examination by interested persons at the Docket Section, Federal Aviation Agency, Room B-316, 1711 New York Avenue NW., Washington 25, D.C. An informal Docket will also be available for examination at the office of the Regional Air Traffic Management Division Chief.

This amendment is proposed under section 307(a) of the Federal Aviation Act of 1958 (72 Stat. 749; 49 U.S.C. 1348).

Issued in Washington, D.C., on February 6, 1961.

CHARLES W. CARMODY,
Chief, Airspace Utilization Division.

[F.R. Doc. 61-1198; Filed, Feb. 10, 1961; 8:46 a.m.]

and to maintain orderly marketing conditions in the respective marketing areas, and

(c) The short time between issuance of this determination and its effective date makes it impractical for rule-making procedure to be completed. The respective orders direct announcement by the market administrator not later than April 15, 1962, of the March indexes employed in the Class I price formulas.

Therefore, good cause exists for making this determination effective on issuance.

Effective date: Upon issuance.

Signed at Washington, D.C., on April 6, 1962.

JOHN P. DUNCAN, JR.,
Assistant Secretary.

[F.R. Doc. 62-3576; Filed, Apr. 11, 1962; 8:50 a.m.]

Title 10—ATOMIC ENERGY

Chapter I—Atomic Energy Commission

PART 100—REACTOR SITE CRITERIA

Pursuant to the Administrative Procedures Act and the Atomic Energy Act of 1954, as amended, the following guide is published as a document subject to codification, to be effective 30 days after publication in the FEDERAL REGISTER.

Statement of considerations. On February 11, 1961, the Atomic Energy Commission published in the FEDERAL REGISTER a notice of proposed rule making that set forth general criteria in the form of guides and factors to be considered in the evaluation of proposed sites for power and testing reactors. The Commission has received many comments from individuals and organizations, including several from foreign countries, reflecting the widespread sensitivity and importance of the subject of site selection for reactors. Formal communications have been received on the published guides, including a proposed comprehensive revision of the guides into an alternate form.

In these communications, there was almost unanimous support of the Commission's proposal to issue guidance in some form on site selections, and acceptance of the basic factors included in the proposed guides, particularly in the proposal to issue exposure dose values which could be used for reference in the evaluation of reactor sites with respect to potential reactor accidents of exceedingly low probability of occurrence.

On the other hand, many features of the proposed guides were singled out for criticism by a large proportion of the correspondents. This was particularly the case for the appendix section of the proposed guides, in which was included an example calculation of environmental distance characteristics for a hypothetical reactor. In this appendix, specific numerical values were employed in the calculations. The choice of these numerical values, in some cases involving simplifying assumptions of highly complex phenomena, represent types of considerations presently applied in site calculations and result in environmental distance parameters in general accord with present siting practice. Nevertheless, these particular numerical values and the use of a single example calculation were widely objected to, basically on the grounds that they presented an aspect of inflexibility to the guides which otherwise appeared to possess considerable flexibility and tended to emphasize unduly the concept of environmental isolation for reactors with minimum possibility being extended for eventual substitution thereof of engineered safeguard.

In consequence of these many comments, criticisms and recommendations, the proposed guides have been rewritten, with incorporation of a number of suggestions for clarification and simplification, and elimination of the numerical values and example calculation formerly constituting the appendix to the guides. In lieu of the appendix, some guidance has been incorporated in the text itself to indicate the considerations that led to establishing the exposure values set forth. However, in recognition of the advantage of example calculations in providing preliminary guidance to application of the principles set forth, the AEC will publish separately in the form of a technical information document a discussion of these calculations.

These guides and the technical information document are intended to reflect past practice and current policy of the Commission of keeping stationary power and test reactors away from densely populated centers. It should be equally understood, however, 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.

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. 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 specification of such a distance requirement would approximately fulfill the desired objectives and reflect a mere 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 some cases where very large cities are involved, the population center distance may have to be greater than those suggested by these guides.

A number of comments received pointed out that AEC siting factors included considerations of population distributions and land use surrounding proposed sites but did not indicate how future population growth might affect sites initially approved. To the extent possible, AEC review of the land use surrounding a proposed site includes considerations of potential residential growth. The guides tend toward requiring sufficient isolation to preclude any immediate problem. In the meantime, operating experience that will be acquired from plants already licensed to operate should provide a more definitive basis for weighing the effectiveness of engineered safeguards versus plant isolation as a public safeguard.

These criteria are based upon a weighing of factors characteristic of conditions in the United States and may not represent the most appropriate procedure nor optimum emphasis on the various interdependent factors involved in selection of sites for reactors in other countries where national needs, resources, policies and other factors may be greatly different.

Sec.

AUTHORITY: §§ 100.1 to 100.11 issued under sec. 103, 68 Stat. 936, sec. 104, 68 Stat. 937, sec. 161, 68 Stat. 948, sec. 182, 68 Stat. 953; 42 U.S.C. 2133, 2134, 2201, 2232.

§ 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 and 115 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. Residence 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 § 50.21(b) or 50.22 of this chapter designed to produce electrical or heat energy.

(e) "Testing reactor" means a "testing facility" as defined in § 50.2 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) Intended 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 use characteristics of the site environs, including the exclusion area, low population zone, and population center distance.

(c) Physical characteristics of the site, including seismology, meteorology, geology and hydrology.

(1) The design for the facility should conform to accepted building codes or standards for areas having equivalent earthquake histories. No facility should be located closer than one-fourth mile from the surface location of a known active earthquake fault.

(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 produce release[1] 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) 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[2] or a total radiation dose in excess of 300 rem[2] 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.

[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 once in a lifetime accidental or emergency dose for radiation workers which, according to NCRP recommendations may be disregarded in the determination of their radiation exposure status (see NBS Handbook 69 dated June 5, 1959). 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, this 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.

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 AEC 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, Division of Licensing and Regulation, U.S. Atomic Energy Commission, Washington 25, D.C.

Dated at Germantown, Md., this 5th day of April 1962.

For the Atomic Energy Commission.

WOODFORD B. MCCOOL,
Secretary.

[F.R. Doc. 62-3523; Filed, Apr. 11, 1962; 8:45 a.m.]

Title 12—BANKS AND BANKING

Chapter II—Federal Reserve System

SUBCHAPTER A—BOARD OF GOVERNORS OF THE FEDERAL RESERVE SYSTEM

[Reg. Q]

PART 217—PAYMENT OF INTEREST ON DEPOSITS

Matured Time Certificate

§ 217.125 Rate of interest on savings deposit created from matured time certificate.

(a) The opinion of the Board of Governors of the Federal Reserve System has been requested by a member bank as to whether a savings deposit may bear a maximum rate of interest of 4 percent from the date of the establishment of the savings deposit by the transfer thereto of funds that have been on deposit for one year as a time certificate.

(b) In § 217.6 it is provided, in part, that the maximum rate of 4 percent may be paid "on that portion of any savings deposit that has remained on deposit for not less than 12 months." This language necessarily implies that the funds must have remained on deposit for 12 months as a savings deposit. To construe the provision otherwise would nullify its purpose. The fact that the depositor was eligible to maintain a savings deposit at all times does not have any significant bearing on the question.

(Sec. 11(i), 38 Stat. 262; 12 U.S.C. 248(i). Interprets or applies secs. 19, 24, 38 Stat. 270, 273, as amended, sec. 8, 48 Stat. 168, as amended; 12 U.S.C. 264(c)(7), 371, 371a, 371b, 461)

Dated at Washington, D.C., this 3d day of April 1962.

BOARD OF GOVERNORS OF THE FEDERAL RESERVE SYSTEM,
[SEAL] MERRITT SHERMAN,
Secretary.

[F.R. Doc. 62-3528; Filed, Apr. 11, 1962; 8:45 a.m.]

[Reg. T]

PART 220—CREDIT BY BROKERS, DEALERS, AND MEMBERS OF NATIONAL SECURITIES EXCHANGES

Special Cash Account

§ 220.117 Exception to 90-day rule in special cash account.

(a) The Board of Governors has recently interpreted certain of the provisions of § 220.4(c)(8), with respect to the withdrawal of proceeds of a sale of stock in a "special cash account" when the stock has been sold out of the account prior to payment for its purchase.

(b) The specific factual situation presented may be summarized as follows:

Customer purchased stock in a special cash account with a member firm on Day 1. On Day 3 customer sold the same stock at a profit. On Day 8 customer delivered his check for the cost of the purchase to the creditor (member firm). On Day 9 the creditor mailed to the customer a check for the proceeds of the sale.

(c) Section 220.4(c)(8) prohibits a creditor, as a general rule, from effecting a purchase of a security in a customer's special cash account if any security has been purchased in that account during the preceding 90 days and has then been sold in the account or delivered out to any broker or dealer without having been previously paid for in full by the customer. One exception to this general rule reads as follows:

* * * The creditor may disregard for the purposes of this subparagraph (§ 220.4(c)(8)) a sale without prior payment provided full cash payment is received within the period described by subparagraph (2) of this paragraph (seven days after the date of purchase) and the customer has not withdrawn the proceeds of sale on or before the day on which such payment (and also final payment of any check received in that connection) is received. * * *

(d) Final payment of customer's check: (1) The first question is: When is the creditor to be regarded as having received "final payment of any check received" in connection with the purchase?

(2) The clear purpose of § 220.4(c)(8) is to prevent the use of the proceeds of sale of a stock by a customer to pay for its purchase—i.e., to prevent him from trading on the creditor's funds by being able to deposit the sale proceeds prior to presentment of his own check to the drawee bank. Thus, when a customer undertakes to pay for a purchase by check, that check does not constitute payment for the purchase, within the language and intent of the above-quoted exception in § 220.4(c)(8), until it has been honored by the drawee bank, indicating the sufficiency of his account to pay the check.

(3) The phrase "final payment of any check" is interpreted as above notwithstanding § 220.6(f), which provides that:

For the purposes of this part (Regulation T), a creditor may, at his option (1) treat the receipt in good faith of any check or draft drawn on a bank which in the ordinary course of business is payable on presentation, * * * as receipt of payment of the amount of such check, draft or order; * * *

This is a general provision substantially the same as language found in section 4(f) of Regulation T as originally promulgated in 1934. The language of the subject exception to the 90-day rule of § 220.4(c)(8), i.e., the exception based expressly on "final payment of any check," was added to the regulation in 1949 by an amendment directed at a specific type of situation. Because the exception is a special, more recent provision, and because § 220.6(f), if controlling, would permit the exception to undermine, to some extent, the effectiveness of the 90-day rule, sound principles of construction require that the phrase "final payment of any check" be given its literal and intended effect.

(4) There is no fixed period of time from the moment of receipt by the payee or of deposit, within which it is certain that any check will be paid by the drawee bank. Therefore, in the rare case where the operation of the subject exception to § 220.4(c)(8) is necessary to avoid application of the 90-day rule, a creditor should ascertain (from his bank of deposit or otherwise) the fact of payment of a customer's check given for the purchase. Having so determined the day of final payment, the creditor can permit withdrawal on any subsequent day.

(e) Mailing as "withdrawal": (1) Also presented is the question whether the mailing to the customer of the creditor's check for the sale proceeds constitutes a withdrawal of such proceeds by the customer at the time of mailing so that, if the check for the sale proceeds is mailed on or before the day on which the customer's check for the purchase is finally paid, the 90-day rule applies. It may be that a check mailed one day will not ordinarily be received by the customer until the next. The Board is of the view, however, that when the check for sale proceeds is issued and released into the mails, the proceeds are to be regarded as withdrawn by the customer; a more liberal interpretation would open a way for circumvention. Accordingly the creditor's check should not be mailed nor the sale proceeds otherwise released to the customer "on or before the day" on which payment for the purchase, including final payment of any check given for such payment, is received by the creditor, as determined in accordance with the principles stated herein.

Relevant excerpts discussing the conversion of the §100.11 criteria to a single TEDE value from the proposed rule 59 FRN 52255 Monday, October 17, 1994, Reactor Site Criteria Including Seismic and Earthquake Engineering Criteria for Nuclear Power Plants and Proposed Denial of Petition From Free Environment, Inc. et al.:

> The Commission has examined the current dose criteria of 25 rem whole body and 300 rem thyroid with the intent of selecting a TEDE numerical value equivalent to the risk implied by the current dose criteria. These risks consist of the risk of developing cancer sometime after the exposure (latent cancer incidence), as well as a delayed risk of cancer fatality (latent cancer fatality). For a dose of 25 rem whole body, the individual risk of latent cancer fatality is estimated to be about 2.5×10^{-2}; the risk of latent cancer incidence is about twice that (using risk coefficients expressed by ICRP Publication 60 and in NUREG/CR-4214). For a dose of 300 rem thyroid, the risk of latent cancer fatality is about 2×10^{-3}; the risk of latent cancer incidence is about a factor of ten higher.
>
> If the risk of latent cancer fatality is selected as the appropriate risk measure to be used, the current dose criteria represent a risk of about 2.7×10^{-2}. Using a risk coefficient of about 10^{-3} per rem, the risk of latent cancer fatality implied by the current dose criteria is equivalent to 27 rem TEDE. (BEIR V estimates a latent cancer fatality risk coefficient of about 5×10^{-4} per rem, if the dose is received over a period of days or more; however, if the exposure period is shorter, such as 2 hours, the risk coefficient is approximately double.)
>
> If latent cancer incidence rather than fatality were used, the current dose criteria would correspond to a value of about 35 rem TEDE.
>
> The Commission is proposing to use the risk of latent cancer fatality as the appropriate risk measure since quantitative health objectives (QHOs) for it have been established in the Commission's Safety Goal policy. Although the current dose criteria are equivalent in risk to 27 rem TEDE, as noted above, the Commission is proposing to use 25 rem TEDE as the dose criterion for plant evaluation purposes, since this value is essentially the same level of risk as the current criteria.
>
> Nevertheless, the Commission is specifically requesting comments on the use of TEDE. Comments are requested on whether the current dose criteria should be modified to utilize the total effective dose equivalent, or TEDE, concept. The Commission is also requesting comments on whether a TEDE value of 25 rem (consistent with latent cancer fatality), or 34 rem (consistent with latent cancer incidence), or some other value should be used. Finally, because the thyroid weighting factor is equal to a value of 0.03, there exists a theoretical possibility that an accidental release composed only of iodine could result in a TEDE less than 25 rem, yet result in a thyroid dose of over 800 rem. Although the Commission believes that the likelihood that an actual accident would release only iodine is highly unlikely, comments are also requested as to whether the dose criterion should also include a "capping" limitation, that is, an additional requirement that the dose to any individual organ not be in excess of some fraction of the total.

The following discussion on the selection of the 25 rem TEDE criterion is from the Final Rule 61 FRN 65157, Wednesday, December 11,1996, Reactor Site Criteria Including Seismic and Earthquake Engineering Criteria for Nuclear Power Plants:

> The Commission considered the current dose criteria of 25 rem whole body and 300 rem thyroid with the intent of selecting a TEDE numerical value equivalent to the risk implied by the current dose criteria. The Commission proposed to use the risk of latent cancer fatality as the appropriate risk measure since quantitative health objectives (QHOs) for it have been established in the Commission's Safety Goal policy. Although the supplementary information in the proposed rule noted that the current dose criteria are equivalent in risk to 27 rem TEDE, the Commission proposed to use 25 rem TEDE as the dose criterion for plant evaluation purposes, since this value is essentially the same level of risk as the current criteria.
>
> However, the Commission specifically requested comments on whether the current dose criteria should be modified to utilize the total effective dose equivalent or TEDE concept, whether a TEDE value of 25 rem (consistent with latent cancer fatality), or 34 rem (consistent with latent cancer incidence), or some other value should be used, and whether the dose criterion should also include a "capping" limitation, that is, an additional requirement that the dose to any individual organ not be in excess of some fraction of the total.
>
> Based on the comments received, there was a general consensus that the use of the TEDE concept was appropriate, and a nearly unanimous opinion that no organ "capping" dose was required, since the TEDE concept provided the appropriate risk weighting for all body organs.
>
> With regard to the value to be used as the dose criterion, a number of comments were received that the proposed value of 25 rem TEDE represented a more restrictive criterion than the current values of 25 rem whole body and 300 rem to the thyroid gland. These commenters noted that the use of organ weighting factors of 1 for the whole body and 0.03 for the thyroid as given in 10 CFR Part 20, would yield a value of 34 rem TEDE for whole body and thyroid doses of 25 and 300 rem, respectively. This is because the organ weighting factors in 10 CFR Part 20 include other effects (e.g., genetic) in addition to latent cancer fatality.
>
> After careful consideration, the Commission has decided to adopt a value of 25 rem TEDE as the dose acceptance criterion for the final rule. The bases for this decision follows. First, the Commission has generally based its regulations on the risk of latent cancer fatality. Although a numerical calculation would lead to a value of 27 rem TEDE, as noted in the discussion that accompanied the proposed rule, the Commission concludes that a value of 25 rem is sufficiently close, and that the use of 27 rather than 25 implies an unwarranted numerical precision. In addition, in terms of occupational dose, Part 20 also permits a once-in-a-lifetime planned special dose of 25 rem TEDE. In addition, EPA guidance sets a limit of 25 rem TEDE for workers performing emergency service such as lifesaving or protection of large populations. While the Commission does not, as noted above, regard this dose value as one that is acceptable for members of the public under accident conditions, it provides a useful perspective with regard to

doses that ought not to be exceeded, even for radiation workers under emergency conditions.

The argument that a criterion of 25 rem TEDE in conjunction with the organ weighting factors of 10 CFR Part 20 for its calculation represents a tightening of the dose criterion, while true in theory, is not true in practice. A review of the dose analyses for operating plants has shown that the thyroid dose limit of 300 rem has been the limiting dose criterion in licensing reviews, and that all operating plants would be able to meet a dose criterion of 25 rem TEDE. Hence, the Commission concludes that, in practice, use of the organ weighting factors of Part 20 together with a dose criterion of 25 rem TEDE, represents a relaxation rather than a tightening of the dose criterion. In adopting this value, the Commission also rejects the view, advanced by some, that the dose calculation is merely a "reference" value that bears no relation to what might be experienced by an actual person in an accident. Although the Commission considers it highly unlikely that an actual person would receive such a dose, because of the conservative and stylized assumptions employed in its calculation, it is conceivable.

PRM-50-87
May 17, 2007

DOCKETED
USNRC
May 25, 2007 (7:51am)

OFFICE OF SECRETARY
RULEMAKINGS AND
ADJUDICATIONS STAFF

Annette L. Vietti-Cook, Secretary
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555-0001

Attention: Rulemakings and Adjudications Staff

Dear Ms. Vietti-Cook:

Pursuant to 10 CFR 2.802, I am submitting the attached petition for rulemaking to the U.S. Nuclear Regulatory Commission (NRC) to request a revision to the regulations specified in 10 CFR 50, Appendix A, "General Design Criteria for Nuclear Power Plants" and in 10 CFR 50.67, "Accident Source Term."

I submit this in the interest of improving the public safety risk from nuclear power plants. I submit this petition as an individual and not on behalf of any group. I have a B.S. in Physics and M.S. in Nuclear Engineering from MIT, and have been performing radiological analyses for nuclear power plants for over 30 years. There are many in the industry, including individuals in the NRC staff, that can attest to my technical capability. The details of the petition are provided in the attachment.

I am providing a copy of this petition to the NRC via e-mail and may be contacted by reply to that e-mail. If desired, I am willing to provide additional information or expand on the alternative solutions provided in the attachment.

Sincerely,

Raymond A Crandall

Attachment

PETITION FOR RULE CHANGE
Raymond Crandall
May 17, 2007

ATTACHMENT

A. CURRENT REGULATIONS

This petition proposes to revise the regulations related to control room habitability at nuclear power plants. The revisions apply to the regulations specified in 10 CFR 50, Appendix A, "General Design Criteria for Nuclear Power Plants" and in 10 CFR 50.67, "Accident Source Term." The following is the current wording of these regulations:

Appendix A to Part 50--General Design Criteria for Nuclear Power Plants

Criterion 19--Control room. A control room shall be provided from which actions can be taken to operate the nuclear power unit safely under normal conditions and to maintain it in a safe condition under accident conditions, including loss-of-coolant accidents. Adequate radiation protection shall be provided to permit access and occupancy of the control room under accident conditions without personnel receiving radiation exposures in excess of 5 rem whole body, or its equivalent to any part of the body, for the duration of the accident. Equipment at appropriate locations outside the control room shall be provided (1) with a design capability for prompt hot shutdown of the reactor, including necessary instrumentation and controls to maintain the unit in a safe condition during hot shutdown, and (2) with a potential capability for subsequent cold shutdown of the reactor through the use of suitable procedures.

Applicants for and holders of construction permits and operating licenses under this part who apply on or after January 10, 1997, applicants for design certifications under part 52 of this chapter who apply on or after January 10, 1997, applicants for and holders of combined licenses under part 52 of this chapter who do not reference a standard design certification, or holders of operating licenses using an alternative source term under § 50.67, shall meet the requirements of this criterion, except that with regard to control room access and occupancy, adequate radiation protection shall be provided to ensure that radiation exposures shall not exceed 0.05 Sv (5 rem) total effective dose equivalent (TEDE) as defined in § 50.2 for the duration of the accident.

§ 50.67 Accident source term.

(a) Applicability. The requirements of this section apply to all holders of operating licenses issued prior to January 10, 1997, and holders of renewed licenses under part 54 of this chapter whose initial operating license was issued prior to January 10, 1997, who seek to revise the current accident source term used in their design basis radiological analyses.

(b) Requirements. (1) A licensee who seeks to revise its current accident source term in design basis radiological consequence analyses shall apply for a license amendment under § 50.90. The application shall contain an evaluation of the consequences of applicable design basis accidents[1] previously analyzed in the safety analysis report.

(2) The NRC may issue the amendment only if the applicant's analysis demonstrates with reasonable assurance that:

(i) An individual located at any point on the boundary of the exclusion area for any 2-hour period following the onset of the postulated fission product release, would not receive a radiation dose in excess of 0.25 Sv (25 rem)[2] total effective dose equivalent (TEDE).

(ii) An individual located at any point on the outer boundary of the low population zone, 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 radiation dose in excess of 0.25 Sv (25 rem) total effective dose equivalent (TEDE).

(iii) Adequate radiation protection is provided to permit access to and occupancy of the control room under accident conditions without personnel receiving radiation exposures in excess of 0.05 Sv (5 rem) total effective dose equivalent (TEDE) for the duration of the accident.

[1] The fission product release assumed for these calculations should be based upon a major accident, hypothesized for purposes of design analyses 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 use of 0.25 Sv (25 rem) TEDE is not intended to imply that this value constitutes an acceptable limit for emergency doses to the public under accident conditions. Rather, this 0.25 Sv (25 rem) TEDE value has been stated in this section as a reference value, which can be used in the evaluation of proposed design basis changes with respect to potential reactor accidents of exceedingly low probability of occurrence and low risk of public exposure to radiation.

B. PROPOSED CHANGE

The proposed change would eliminate the specific criteria related to the radiological doses for control room habitability. The revised regulations would read as follows:

Appendix A to Part 50--General Design Criteria for Nuclear Power Plants

Criterion 19--Control room. A control room shall be provided from which actions can be taken to operate the nuclear power unit safely under normal conditions and to maintain it in a safe

condition under accident conditions, including loss-of-coolant accidents. Equipment at appropriate locations outside the control room shall be provided (1) with a design capability for prompt hot shutdown of the reactor, including necessary instrumentation and controls to maintain the unit in a safe condition during hot shutdown, and (2) with a potential capability for subsequent cold shutdown of the reactor through the use of suitable procedures.

§ 50.67 Accident source term.

(a) Applicability. The requirements of this section apply to all holders of operating licenses issued prior to January 10, 1997, and holders of renewed licenses under part 54 of this chapter whose initial operating license was issued prior to January 10, 1997, who seek to revise the current accident source term used in their design basis radiological analyses.

(b) Requirements. (1) A licensee who seeks to revise its current accident source term in design basis radiological consequence analyses shall apply for a license amendment under § 50.90. The application shall contain an evaluation of the consequences of applicable design basis accidents[1] previously analyzed in the safety analysis report.

(2) The NRC may issue the amendment only if the applicant's analysis demonstrates with reasonable assurance that:

(i) An individual located at any point on the boundary of the exclusion area for any 2-hour period following the onset of the postulated fission product release, would not receive a radiation dose in excess of 0.25 Sv (25 rem)[2] total effective dose equivalent (TEDE).

(ii) An individual located at any point on the outer boundary of the low population zone, 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 radiation dose in excess of 0.25 Sv (25 rem) total effective dose equivalent (TEDE).

[1] The fission product release assumed for these calculations should be based upon a major accident, hypothesized for purposes of design analyses 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 use of 0.25 Sv (25 rem) TEDE is not intended to imply that this value constitutes an acceptable limit for emergency doses to the public under accident conditions. Rather, this 0.25 Sv (25 rem) TEDE value has been stated in this section as a reference value, which can be used in the evaluation of proposed design basis changes with respect to potential reactor accidents of exceedingly low probability of occurrence and low risk of public exposure to radiation.

C. BASIS FOR CHANGE - SUMMARY

This section summarizes the basis for the proposed change. More detailed justifications for the statements in this summary are provided in Section D.

The current regulations provide specific dose criteria for demonstrating the acceptability of the design of the control room for radiological release events. The existence of specific numeric acceptance criteria mandates that the acceptability of the design be based on deterministic radiological dose analyses performed by the licensee and reviewed by the NRC staff. NRC Regulatory Guides and Standard Review Plans provide the methodologies to be used to perform these dose analyses. Many of the site-specific input assumptions used in these dose analyses (e.g., control room inleakage rates and control room ventilation filter efficiencies) are incorporated into the licensees Technical Specifications.

The use of this deterministic dose analysis methodology and associated regulatory process has resulted in the following negative safety consequences:

1. Control room designs that are not optimum for ensuring continued control room habitability. Current designs required in order to meet the current dose methodology criteria may actually increase the probability of having to evacuate the control room compared to establishing the design based on good engineering principles.
2. Site procedures for mitigation of the dose consequences to control room personnel that are not optimum for ensuring control room habitability. The procedures designed to ensure consistency with the dose analysis assumptions are inconsistent with more effective mitigation strategies.
3. Unnecessary challenges to safety systems, such as increased challenges to the Emergency Diesel Generators if control room ventilation system fans are loaded on the diesels early in the accident to meet analysis assumptions.
4. Technical Specifications Action Statement requirements that result in a net increase in the risk to the public. This specifically refers to Technical Specifications that require a plant shutdown for failure to meet a control room dose analysis input assumption.
5. Technical Specifications Surveillance requirements that cannot be cost-justified based on the risk-significance. This results in the required expenditure of resources that could be used on risk-significant improvements.

The proposed rule change would eliminate the specific radiological dose acceptance criteria from the regulation. This would eliminate the need for the deterministic dose analysis and the associated regulatory process associated with that methodology, such as the Technical Specifications imposed to ensure compliance with the methodology. The proposed rule change does not eliminate the requirement for the control room to be designed to ensure safe conditions under accident conditions, including radiological accidents. Alternative methods to ensure appropriate control room designs are provided in Section E. These alternative approaches would eliminate the safety concerns listed above associated with the current regulation.

D. TECHNICAL JUSTIFICATIONS

1. Current designs are not optimum

The current dose analysis criteria are based on a set of very low probability assumptions and on acceptance criteria that are inconsistent with the ultimate goal of the regulation, which is to ensure the control room operators can remain in the control room to mitigate the consequences of the accident. As a result, some common designs installed to ensure compliance with the existing criteria, such as a filtered air-intake pressurization design, actually increase the probability that the control room will require evacuation.

Consider the following facts. In the currently prescribed methodology, the controlling dose pathway is typically the thyroid dose from radioiodine. However, in reality, the control room would likely not be evacuated based on radioiodine levels in the control room. Should high levels of radioiodine enter the control room, the use of KI and/or respiratory protection would allow the operator to remain in the control room, with an acceptably low thyroid dose. However, the control room may have to be evacuated due to the whole body dose rates from noble gases. It would not be possible to shield the operator from the gamma radiation emanating from a cloud of radioactive noble gases that entered the control room. The probability of a release of large quantities of noble gases is much greater than the probability of a release of large quantities of iodine. Due to the many removal mechanisms available for iodines (e.g., sprays, filters, settling, plateout), many scenarios, such as the TMI accident, would result in a significant noble gas release with negligible iodine releases. It is highly improbable to have a high iodine release without a simultaneous high noble gas release.

Since the primary objective of control room habitability is to ensure continuous occupancy of the control room, then the primary focus should be the minimization of the whole body dose from noble gases. Unfortunately, the low probability scenarios and conservative assumptions chosen for the current dose analysis methodologies make the thyroid dose limiting in these analyses. Therefore, design and operational criteria are established to minimize the thyroid dose. This occurs at the expense of increasing the whole body dose. For example, a pressurized control room design continuously draws in outside air through filters that remove iodine in order to pressurize the control room and minimize the inleakage of unfiltered iodine. The filters have no effect on radioactive noble gases. The consequence is that more radioactive noble gases will be drawn into the control room than would have leaked into the control room had the ventilation system been simply isolated upon detection of a radioactive plume, with no filtered makeup. Therefore, if noble gas release rates were higher than assumed, the chances of having to evacuate the control room due to high whole body dose rates increases by pressurizing rather than isolating the control room.

Isolating, rather than pressurizing the control room, may increase the iodine concentration in the control room, but the dose from the increased iodine concentration can be mitigated through the use of KI or respiratory protection. The NRC staff does not allow these mitigating techniques for radiological releases to be used in design analyses. This is inconsistent with the fact that credit for respiratory protection is allowed in control room habitability toxic gas release evaluations.

2. Procedures are not optimized

Dose analysis methodologies are simplified in order make the analysis manageable. The conditions analyzed represent one hypothetical set of conditions and assumptions. Due to the simplifications, some of the assumptions are actually impossible. There are thousands of other possible scenarios, most of which are more likely than the default hypothetical scenario. Procedures for dose mitigation however must be consistent with the licensing basis hypothetical analysis. Such procedures may not be the optimum mitigation strategy for the more likely conditions.

This is best illustrated by example. Placing the control room in a purge mode when outside air at the intake has no airborne radioactivity due to a change in the wind direction of the plume would be the most effective means of reducing the control room operator dose. Purging removes both the iodines and the noble gases that are in the control room. Recirculation through the filters only removes the iodines, and is typically at a lower flow rate than the purge mode.

Due to simplifications in the design basis analysis, the outside air concentration at the control room intake is never zero over the entire 30-day period of the assumed release. In reality, the plume should only be blowing from the release point towards the control room approximately 10% of the time over the long term. The control room dose models do not model dispersion as a period during the day with higher concentrations while the plume is blowing towards the control room and then a period of zero concentration for the remainder of the day. The analysis methods simplify this effect by assuming a lower concentration is present continuously. Since there is always an outside concentration in the analysis, terminating the recirculation mode and initiating a purge mode would increase the calculated dose. If procedures were revised to incorporate a purge mode strategy, such procedures would result in a calculated increase in consequences in the simplistic design basis analysis. Therefore, such mitigation strategies are often not proceduralized since they would be inconsistent with the current regulatory practice of evaluating the effectiveness of the procedure based on the analysis of one hypothetical set of conditions.

It is possible that the emergency response organization would implement a strategy for purging during an actual event, but it would be more likely if such a strategy were already proceduralized. Mitigating strategies should be based on overall risk reduction, which would invoke strategies for the more likely conditions. Mitigating strategies should not be based on one set of fixed hypothetical unlikely conditions, but that is what happens when the regulation requires a deterministic dose analysis.

3. Challenges to safety systems

This is similar to the discussion on procedures but is related to design features that unnecessarily challenge other safety systems. In many cases, design requirements are imposed to ensure the assumptions of the dose analysis are met. Not meeting the assumptions for the specified set of hypothetical conditions may result in the inability to

meet the acceptance criteria. However, from an overall risk perspective, these design requirements may not be optimum.

For example, one common design requirement is that the normal control room ventilation isolate upon a Safety Injection or Containment Isolation signal. This is necessary because analysis simplifications place the plume at the control room intake immediately following the assumed LOCA. Hence prompt isolation is required to avoid an initial intake and meet the dose limits. It is more likely that there will be no radioactive plume by the control room at time zero of the accident. The fuel will likely not fail, the containment will likely not leak, and/or the plume will be blowing another direction. Isolating normal control room ventilation has negative consequences in regard to control room temperature and humidity control, which in tum could have negative effects on equipment or operators in the control room. It would be more logical to delay control room isolation until radioactivity is detected in the control room or until it is known that there is a radioactive plume blowing in the general direction of the control room.

As another example, in order to meet the dose limits in the analysis world where the plume is instantaneously at the control room intake, the control room recirculation fans must often be loaded quickly on the emergency diesel generators following an assumed loss of offsite power. This adds additional challenges to the diesel generator, or could result in a loading scheme where systems that may be more beneficial from an overall safety risk are loaded later.

Mitigating design strategies should be based on overall risk reduction, which would favor designs for the more likely conditions. Mitigating design strategies should not be based on one set of fixed hypothetical unlikely conditions, but that is what happens when the regulation requires a deterministic dose analysis.

4. Inappropriate Technical Specifications Action Statements

To understand the basis behind this point and the subsequent point on surveillance requirements, it is necessary to provide some insight into the performance of design basis radiological dose analyses, as radiological analyses are different from other types of engineering calculations.

Most engineering analyses involve some amount of uncertainty. For non-radiological analyses, even with this uncertainty, the results still reasonably match what can be expected in a real event. For example, for the thermal-hydraulic analyses for an assumed LOCA event, numerous assumptions go into the analysis to demonstrate that fuel damage will not occur due to overheating. For some assumptions, such as the heat removal capability of the metal mass, the conservatism is built into the model. For other assumptions, such as the temperature of the safety injection water, or the flow rate of the safety injection pump, the uncertainty is limited by specifying an acceptable value for such a parameter in the Technical Specifications. The analysis uses the most conservative of the Technical Specifications allowable values. The results using this

approach are generally conservative, but will likely be accurate within an order of magnitude. For example, if a calculation determined that a safety injection flow of 400 gpm ensures fuel integrity, if in a real event, the pump could only deliver 200 gpm, fuel damage would likely not occur due to the conservatisms in the analysis. However, if the pump could only provide 40 gpm, fuel damage may occur. There are numerous other engineering examples where analysis assumptions must be correct within a small factor or serious consequences could occur, such as a bridge collapse or airliner crash. This has resulted in the need to treat each significant input assumption with some importance. Thus the Technical Specifications requirements for a safety injection system that cannot meet its design requirements will impose a shutdown requirement.

Unfortunately this same philosophy has carried over into the treatment of assumptions used in radiological analyses. The failure to meet specified input assumptions would result in declaring the habitability system inoperable and an eventual shutdown. Due to the large number of assumptions in a radiological analysis, and the large uncertainties in most of them as described below, each individual assumption is essentially meaningless, as is the final result. Yet the Technical Specifications treat each assumption as if the failure to meet that assumption will result in unacceptable consequences and hence shutdown requirements are imposed.

Since the LOCA is typically the limiting accident for control room habitability design, the following discussion is based on a large break LOCA. The radiological analysis requires multiple inputs. First is the source term, which is the amount of radioactivity released from the core and available for release to the environment. This assumption can vary by nine orders of magnitude. The release can be the 1 Curie of radioactivity in the reactor coolant assuming no fuel damage, which is the expected case; or, it can be the lx109 Curies in the core if we assume core melt. In the philosophy of defense in depth, since the analyses are being used to design dose mitigating features, it is appropriate to assume the core melt source term of $1x10^9$ Curies, since such a source term is possible. Therefore, for this assumption, it is reasonable to specify an assumption at the high end of the uncertainty. However, it is not just the total curies released that is an important source term consideration. The calculated and actual dose will depend significantly (many orders of magnitude variation) on the nuclide mix of the release, which in itself is highly dependent on the operating history, the decay time since reactor shutdown, and the fraction of particulate nuclides that become airborne during the event. The dose is also highly dependent on the chemical form of the source term. For example, the ability to remove iodine from the air prior to release depends on whether it is in a gaseous form or particulate form.

The next set of uncertainty is related to the removal mechanism for the various nuclides. For example, for iodine, which in the design basis analyses is the most significant dose nuclide, there are numerous removal mechanisms. This includes filtration, spray removal, deposition, and plate out. Each of these removal mechanisms is typically modeled very conservatively rather than the use of best-estimate or nominal values. In most cases, other removal mechanisms, such as diffusiophoresis or non-safety grade

filters are not even modeled, even though they will provide a significant reduction in the release.

The next set of uncertainties is related to the release pathway and the motive forces necessary to cause a release via that pathway. The actual leak rate from the containment can vary by many orders of magnitude, as can the leak rate from systems recirculating containment sump water outside the containment. Containment pressure, which provides the motive force for release from the containment atmosphere can vary significantly. The temperature of the containment sump water at the leak location outside containment significantly impacts the fraction of activity that becomes airborne. Again, each of these assumptions is typically taken near the high end of conservatism. For example, the containment pressure is assumed to remain at the maximum calculated pressure for 24 hours, even though it will likely rapidly decrease within minutes.

The next set of uncertainties is related to atmospheric dispersion. The airborne concentration of the released radioactivity at a downwind receptor location can vary by approximately six orders of magnitude. The concentration with high wind speeds and unstable atmospheric conditions can be a million times less than a condition with low wind speed and stable atmospheric conditions. The design basis analysis rules require the assumption of low wind speed, very stable conditions. The design bases analyses also require that the wind direction be directly towards the control room and directly towards the closest site boundary during the period of highest releases. The analyses also assume that there is a person at that closest site boundary location; otherwise the dose at that location would be zero.

There are many other uncertainties, such as in the dose modeling, which depends on the breathing rate of the individual, the size of the individual, biological removal mechanisms, etc., but these uncertainties are small compared to the many assumptions that have orders-of-magnitude uncertainties.

Given so many assumptions with orders-of-magnitude uncertainties, and given that the analysis requirements typically specify that each uncertain assumption be at the high end of the uncertainty for conservatism, it renders the final result meaningless. The combined probability of all assumptions being at the high end of uncertainty is so small that the design basis event is incredible and will not match reality. This makes each individual assumption meaningless in regard to predicting actual results. For example, if in a real event the ventilation filter efficiency is somewhat less than the assumed value in the analysis, the small reduction in iodine removal capability of the filters is likely more than compensated for by other iodine removal mechanisms not taken credit for, or by lower containment leak rates than assumed, or higher wind speeds than assumed.

The dose from the Three Mile Island accident came predominantly from pathways that aren't modeled (sump water pumped back to radwaste and letdown system leakage). The dose from modeled pathways was insignificant. Iodine releases were insignificant, even though it is the most significant dose nuclide in the analysis. Therefore, TMI LOCA

dose analysis input assumptions had no significance in predicting the actual consequences of the event.

Yet, the principles from other types of engineering analyses are applied to these dose analysis input assumptions. They are incorporated into the Technical Specifications with shutdown requirements based on the assumption that not meeting these assumptions implies unacceptable consequences.

The above discussion could provide a reasonable argument for removal of shutdown requirements for dose mitigating systems that are directly related to public dose. However, that is not the subject of this proposed rule change. This proposed rule change is related to the analyses and Technical Specifications for control room habitability. For control room habitability, the analyses assumptions and results are even further removed from having any significance. First, there is no direct public impact from not meeting the control room habitability system requirements. The control room inleakage rate or control room filter efficiencies are not factors in the public dose analyses. Second, if an input assumption could not be met, and by some small chance all the other conservative assumptions were true such that the potential dose to the control room operator would exceed the acceptable limits, this dose can be easily mitigated by simply providing the operator with KI. Third, the dose limit itself is overly restrictive. Why should the public be allowed to receive 25 REM TEDE and the control room operator be limited to 5 REM? There is no health consequence to a dose of 25 REM, and the EPA protective action guidelines would allow such a dose for control room operator functions. In the past, in an attempt to find some safety significance to the control room habitability requirements, the NRC staff has stated that the operators may not feel adequately protected to perform their function if the plant conditions and design analyses did not demonstrate that the 5 REM limit could be met. The control room operator is a trained nuclear professional, dedicated to the protection of public safety, and would be willing to receive a dose higher than 5 REM to mitigate an accident.

Therefore, the potential indirect impact on public safety of having to evacuate the control room can be easily avoided, regardless of the control room habitability system status. There is insignificant safety significance to the Technical Specifications associated with control room habitability and yet there are shutdown requirements.

In the past, on numerous occasions, the NRC has specified that the inability to meet the assumptions or criteria of control room habitability analyses has low safety significance. This has been stated in the interim and final closure for various plants of NUREG-0737, TMI lessons learned criterion III.D.3.4, Control Room Habitability. It has also been stated recently when various plants have measured inleakage values well in excess of the analysis assumed values. The primary basis for the low safety significance was typically the existence of simple mitigating actions such as the issuance of KI tablets that ensure the continued occupancy of the control room. This low safety significance has been used to justify continued operation.

In order to evaluate the net public safety risk associated with these Technical Specifications shutdown requirements, one must consider the small but quantifiable public risks associated with the shutdown of a nuclear power plant. These include, but are not limited to:

- The risk associated with bringing the plant through a transient and another thermal cycle
- The airborne pollutants released by the fossil units required to operate to make up for the lost power
- The potential for challenging the stability of the electric power grid, with the public risk associated with the possibility of rolling blackouts or brownouts, or under the worst conditions of grid instability, the potential for a loss of offsite power event at multiple nuclear power plants.

Although these public risks associated with a shutdown are small, given the insignificant risk associated with not meeting the control room habitability system requirements, the shutdown requirement is actually increasing the net public risk. Imagine a scenario where a nuclear power plant had to shutdown due to a failed control room habitability surveillance and this unexpected shutdown challenged the power grid to where rolling blackouts were required. Public deaths were then attributed to carbon monoxide poisoning from use of an alternate heat source. Compare that consequence with the consequence of the failed control room habitability surveillance, which was a small increase in the potential to have to provide the control room operator with KI, assuming that a one-in-a-million probability accident occurs.

The shutdown requirement for these surveillances needs to be eliminated. The shutdown requirement is only imposed as a "matter of compliance," which stems from the manner in which the input assumptions are treated when using deterministic calculations.

5. Unjustified Technical Specification Surveillances

Section D.4 demonstrated that the individual input assumptions for radiological dose analyses have no significance in predicting reality or the acceptability of results. Even if actual conditions were such that one of the assumptions was non-conservative by a couple orders of magnitude, the ultimate result (in this case habitability of the control room) would still be acceptable due to the significant conservatisms in the other assumptions and the simplicity of effective mitigating actions such as the use of KI.

The lack of any safety significance to the input assumption should impact the effort that is required to demonstrate the accuracy and conservatism of a specific input assumption. Most control room habitability surveillances can be performed relatively easily with minimal resources. However, over the past seven years, licensees have been required to demonstrate the accuracy of the assumption on unfiltered inleakage using a testing method that cannot be cost-justified. The required tracer-gas testing method costs approximately $100,000 per test. During 2007, most licensees will be

required to incorporate the routine performance of this test into their Technical Specifications. The incorporation of this new requirement was imposed without any implementation of the back-fit rule, based on the determination that performance of this test was a "matter of compliance." This "matter of compliance" stems from the manner in which the input assumptions are treated when using deterministic calculations.

The tracer gas testing performed to date did demonstrate that the inleakage values assumed in the analyses were typically non-conservative. The tracer gas tests also demonstrated that surveillances performed to date, such as a control room pressurization tests, failed to demonstrate this non-conservatism. There were a number of lessons learned from the performance of these tests as to the sources of the unfiltered inleakage (e.g., from leaks into the negative pressure sections of ductwork of the control room ventilation system if located outside the habitability envelope).

Most of the results were within an order of magnitude of the assumed inleakage. Based on the discussion above, the tests therefore demonstrated that this is one of the least uncertain assumptions. Being within an order of magnitude, even if non-conservative, is more than compensated for by the conservatism in many other assumptions. Additionally, the consequences of a higher unfiltered inleakage are easily compensated for through the use of KI. These facts were used to demonstrate that even for those few licensees where the results were greater than an order of magnitude non-conservative, that there was no safety significance and continued operation was justified.

If the actual results of the test have no safety significance, then the significant cost cannot be justified. The optional station improvements that may be postponed due to the reduction in the budget of $100,000 for this test would likely be more beneficial in overall safety and reliability. Therefore, the required performance of this test could have a net negative safety consequence. It is proposed that performance of the previous surveillances, such as a pressurization test, along with incorporation of the lessons learned from the tracer gas testing into an effective preventative maintenance program for boundary integrity is sufficient. It provides a cost-justified approach to ensure that there are no significant failures of the control room habitability boundary and hence that there would be an insignificant potential to have to evacuate the control room.

E. PROPOSED ALTERNATIVES

The preferred option is as recommended in Section B, which is a rule change to eliminate the specific radiological criteria for control room habitability. This would then result in the ability to revise the industry guidelines to eliminate the specified guidelines for performing deterministic dose analyses. This would result in the ability to eliminate all of the negative safety consequences discussed above that result from such an approach.

The current guidelines could be replaced with guidelines based on good engineering principles that would ensure that the control room remains habitable under most postulated conditions.
As an example, the guidance could include requirements such as:

- The control room ventilation system should isolate on the detection of high radiation or toxic gas intake.
- The control room should have a minimum of one foot of concrete shielding (or equivalent) on all surfaces.
- Self Contained Breathing Apparatus (SCBA' s) and potassium iodide (KI) tablets should be readily available for operator use. Operators should maintain training in SCBA's.
- Procedural controls to maintain a low leakage boundary, such as preventative maintenance/routine inspection of door seals and dampers should be implemented.
- Procedures should be developed to ensure control room purging is considered when the outside concentration is less than the inside concentration.
- Existing emergency filtration systems should be maintained to practical performance criteria

The current Technical Specifications for system performance would be eliminated. The Administrative Section of the Technical Specifications could include a requirement to have a Control Room Habitability Program. A guidance document (e.g. - Regulatory Guide or endorsement of an NEI guidance document) could be written to specify the aspects required in such a program.

Given the low public risk significance of being outside these design guidelines established in a licensee's Control Room Habitability Program, a plant shutdown would not be required if outside the guidelines. Rather, the program could specify that timely actions should be taken to return the plant to within the guidelines. If not completed within 30 days, a Special Report would be sent to the NRC with a justification for continued operation and proposed schedule for meeting the guidelines.

Removing the specific dose criteria from the rule would not eliminate the need to perform quantitative analyses if required to demonstrate the acceptability for certain conditions. For example, the guideline above for one foot of concrete shielding could be expanded to require a quantitative assessment of the shielding adequacy if a significant radiation source (e.g., a post- accident release filtration system) is located immediately outside the control room wall. The current rule has no specific quantitative limits for toxic gases, yet the guidelines require quantitative analyses for toxic gas habitability assessments under certain conditions.

As an alternative to total removal of dose guidelines from the rule, most of the concerns noted above could be resolved if the dose criteria were based solely on the whole body dose from noble gases, which is likely the only possible dose impact that may result in control room evacuation. As another option, most of the concerns would be resolved if credit for SCBA's and/or KI was allowed in the analysis of the dose from iodines and particulates. These options would need to be accompanied by changes in the guidelines, such as a revision to generic Technical Specifications to eliminate shutdown requirements for failure to meet control room habitability system requirements in order for the benefits noted above to be realized. If one of these alternatives is preferred, I would be happy to provide additional input and details on how such options can maximize public safety.

F. CONCLUSION

It should be noted that many of the points in my technical discussion above have been presented to NRC staff in various industry forums, but not as formally and not all at one time. However, the current resulting practices and requirements have prevailed based on NRC staff statements that "It's a matter of compliance," rather than on logical choices to improve overall public risk or impose cost-beneficial requirements. If the current rule that requires the use of deterministic dose consequences is what has resulted in this type of decision-making, then it is time to revise that rule.

A rule change that eliminates the approach of using deterministic control room habitability dose analyses to establish requirements would have a number of benefits that would reduce overall public safety risk. Such benefits would result from improved designs, improved procedures, reduced challenges to other safety systems, improved Technical Specifications action requirements and improved surveillance requirements. It is recommended that the NRC implement the rule change proposed in Section B in the interest of public safety.

Additionally, recognizing the time it requires to implement a rule change, it is also recommended that more timely actions be taken within the requirements of the current rule. For example, such actions would include Technical Specifications changes to eliminate shutdown requirements for control room habitability.

Proposed Rules

Federal Register
Vol. 74, No. 15
Monday, January 26, 2009

This section of the FEDERAL REGISTER contains notices to the public of the proposed issuance of rules and regulations. The purpose of these notices is to give interested persons an opportunity to participate in the rule making prior to the adoption of the final rules.

NUCLEAR REGULATORY COMMISSION

10 CFR Part 50

[NRC–2007–0016; PRM–50–87]

Raymond A. Crandall; Denial of Petition for Rulemaking

AGENCY: Nuclear Regulatory Commission.

ACTION: Denial of petition for rulemaking.

SUMMARY: The Nuclear Regulatory Commission (NRC) is denying the petition for rulemaking (PRM) filed by Mr. Raymond A. Crandall on May 17, 2007, and docketed on June 22, 2007 (Docket No. PRM–50–87). In his petition, the petitioner requested that the NRC amend the regulations that govern domestic licensing of production and utilization facilities to eliminate the specific criteria related to the radiological doses for control room habitability at nuclear power plants. The petitioner stated that the current deterministic radiological dose requirements for control room habitability have resulted in several negative safety consequences, including an increased risk to public safety. He requested that the NRC delete the 5 rem whole body dose limit and the 0.05 sievert (Sv) (5 rem) total effective dose equivalent (TEDE) limit specified in the current regulations.

DATES: The docket for PRM–50–87 is closed as of January 26, 2009.

ADDRESSES: Publicly available documents related to this petition, including the PRM and the NRC's letter of denial to the petitioner may be viewed using the following methods:

Federal e-Rulemaking Portal: Go to *http://www.regulations.gov* and search for documents related to this PRM filed under docket ID NRC–2007–0016.

NRC's Public Document Room (PDR): The public may examine publicly available documents and have them copied for a fee at the NRC's PDR, Public File Area O–1 F21, One White Flint North, 11555 Rockville Pike, Rockville, Maryland.

NRC's Agencywide Document Access and Management System (ADAMS): Publicly available documents created or received at the NRC are available electronically via the NRC's Electronic Reading Room at *http://www.nrc.gov/NRC/reading-rm/adams.html*. From this page, the public can gain entry into ADAMS, which provides text and image files of the NRC's public documents. If you do not have access to ADAMS or have any problems in accessing the documents located in ADAMS, contact the NRC PDR Reference staff at 1–800–397–4209, or 301–415–4737, or by e-mail to *PDR.resource@nrc.gov*.

FOR FURTHER INFORMATION CONTACT: A. Jason Lising, Office of Nuclear Reactor Regulation, U.S. Nuclear Regulatory Commission, Washington, DC 20555–0001, telephone: (301) 415–3220, or toll-free: 800–368–5642; e-mail: *Jason.Lising@nrc.gov*.

SUPPLEMENTARY INFORMATION:

I. Background

On May 17, 2007, the NRC received a PRM from Raymond A. Crandall (ADAMS Accession No. ML071490250); the PRM was docketed by the NRC as PRM–50–87. The petitioner requested that the NRC amend Title 10 of the *Code of Federal Regulations* Part 50 (10 CFR Part 50), ''Domestic Licensing of Production and Utilization Facilities'' to remove the specific criteria related to the radiological doses for control room habitability at nuclear power plants from 10 CFR 50.67, ''Accident source term,'' and General Design Criterion (GDC) 19, ''Control room,'' in Appendix A, ''General Design Criteria for Nuclear Power Plants,'' to 10 CFR Part 50. The NRC published a notice of receipt and request for public comment in the **Federal Register** on July 12, 2007 (72 FR 38030). The 75-day public comment period ended on September 25, 2007.

The petitioner noted that the current regulations provide specific dose criteria for demonstrating the acceptability of the control room design during radiological release events. These criteria are based on deterministic radiological dose analyses performed by the licensee and reviewed by the NRC. NRC regulatory guides and standard review plans provide acceptable methodologies that can be used by licensees to perform dose analyses, which are then incorporated, as appropriate, into the licensing basis for the licensee's facility. The petitioner stated that the deterministic dose analysis methodology and associated regulatory process result in several negative safety consequences:

(1) Current Designs Not Optimum

''Control room designs that are not optimum for ensuring continued control room habitability. Current designs required in order to meet the current dose methodology criteria may actually increase the probability of having to evacuate the control room compared to establishing the design based on good engineering principles.''

(2) Procedures Not Optimized

''Site procedures for mitigation of the dose consequences to control room personnel that are not optimum for ensuring control room habitability. The procedures designed to ensure consistency with the dose analysis assumptions are inconsistent with more effective mitigation strategies.''

(3) Challenges to Safety Systems

''Unnecessary challenges to safety systems, such as increased challenges to the Emergency Diesel Generators if control room ventilation system fans are loaded on the diesels early in the accident to meet analysis assumptions.''

(4) Inappropriate Technical Specification (TS) Action Statements

''Technical Specifications Action Statement requirements that result in a net increase in the risk to the public. This specifically refers to Technical Specifications that require a plant shutdown for failure to meet a control room dose analysis input assumption.''

(5) Unjustified Technical Specification Surveillances

''Technical Specifications Surveillance requirements that cannot be cost-justified based on the risk-significance. This results in the required expenditure of resources that could be used on risk-significant improvements.''

The petitioner suggested amendments that would eliminate the specific radiological dose acceptance criteria and, thereby, the need for deterministic dose analyses and the associated regulatory processes, including the need for applicable TSs. He stated that the

proposed changes would not eliminate the requirement for the control room to be designed to ensure safe conditions under accident conditions, but it would address his safety concerns with the current regulations.

II. Petitioner's Request

In PRM–50–87 the petitioner requested that the NRC take the following actions:

1. Revise the regulations related to control room habitability at nuclear power plants by deleting the following sentences from GDC 19:

"Adequate radiation protection shall be provided to permit access and occupancy of the control room under accident conditions without personnel receiving radiation exposures in excess of 5 rem whole body, or its equivalent to any part of the body, for the duration of the accident. Applicants for and holders of construction permits and operating licenses under this part who apply on or after January 10, 1997, applicants for design certifications under part 52 of this chapter who apply on or after January 10, 1997, applicants for and holders of combined licenses under part 52 of this chapter who do not reference a standard design certification, or holders of operating licenses using an alternative source term under § 50.67, shall meet the requirements of this criterion, except that with regard to control room access and occupancy, adequate radiation protection shall be provided to ensure that radiation exposures shall not exceed 0.05 Sv (5 rem) total effective dose equivalent (TEDE) as defined in § 50.2 for the duration of the accident."

2. Revise the regulations related to control room habitability at nuclear power plants to delete from paragraph (b)(2)(iii) in 10 CFR 50.67 this language:

"Adequate radiation protection is provided to permit access to and occupancy of the control room under accident conditions without personnel receiving radiation exposures in excess of 0.05 Sv (5 rem) total effective dose equivalent (TEDE) for the duration of the accident."

III. Reasons for Denial

1. General

The NRC has reviewed Mr. Raymond Crandall's petition and has determined that it does not provide adequate justification to remove the control room radiological dose acceptance criteria from NRC regulations. The NRC does not agree with the petitioner's assertion that the control room radiological dose acceptance criteria have resulted in negative safety consequences.

Performance-based regulations, such as § 50.67 and Appendix A to 10 CFR Part 50, do not provide prescriptive requirements and, therefore, do not require licensees to use specific designs or methodologies to comply with the regulations. The NRC, however, does provide regulatory guidance to licensees that includes acceptable designs and methodologies for demonstrating compliance with the regulations. The use of the guidance is optional, and licensees are free to propose alternative means of complying with the NRC's regulations.

Design-basis dose consequence analyses are intentionally based upon conservative assumptions and are intended to model the potential hazards that would result from any credible accident, not necessarily the most probable accident. As stated in footnotes to 10 CFR 100.11, "Determination of exclusion area, low population zone, and population center distance," and 10 CFR 50.67, "Accident source term," "[t]he 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."

The performance-based control room dose criterion is designed to maintain an acceptable level of control room habitability even under the maximum credible accident scenario. The NRC has determined that providing an acceptable level of control room habitability for design-basis events is necessary to provide reasonable assurance that the control room will continue to be effectively manned and operated to mitigate the effects of the accident and protect public health and safety. Meeting or exceeding the design-basis control room dose limit would not impose an immediate evacuation requirement on the control room operators. Moreover, by removing the 5 rem acceptance criterion, a regulatory basis for the acceptance of the radiological protection aspects of control room designs would no longer exist and would not support the Commission's policy regarding performance-based regulations.

The conservative assumptions used in design-basis dose consequence analyses need not and should not form the basis for restricting actions described in emergency operating procedures. These procedures are designed to ensure that during an accident all available means are used to assess actual radiological conditions and to maintain emergency worker doses As Low As Reasonably Achievable (ALARA), as required by 10 CFR Part 20, "Standards For Protection Against Radiation." Additionally, no NRC regulations, including 10 CFR Part 20, "Standards for Protection Against Radiation," require evacuation of the control room when the design-basis control room dose limit is exceeded. Emergency operating procedures include guidance for controlling doses to workers under emergency conditions. This guidance would be applicable in the unlikely event that control room doses were projected to exceed the design-basis dose limit during an actual emergency.

2. NRC Staff Responses to the Petitioner's Assertions

A. Current Designs Are Not Optimum

1. The petitioner stated that because the primary objective of control room habitability is to ensure continuous occupancy, the primary focus should be on minimizing whole body doses from noble gases. He stated that some common control room designs, such as the filtered air intake pressurization design, focus on compliance with existing dose criteria. He concluded that the current requirements and operational criteria focus on minimizing the thyroid dose at the expense of increasing the whole body dose from noble gases which increases the probability that the control room will require evacuation.

The NRC reviewed the petitioner's concern regarding the increase in whole body dose from noble gases, which he believes results from the intentional intake of filtered air into the control room under design-basis accident (DBA) conditions. The NRC agrees that a relatively small increase in whole body dose due to noble gases may result from the intake of filtered air into the control room. However, this small increase in dose would not increase the probability of a control room evacuation. Therefore, operators would be able to monitor plant indications and take appropriate accident mitigating actions from the control room, and there would be no increase in risk to public health and safety. The NRC's conclusion is based on a review of several existing DBA control room dose analyses that determined the impact on whole body dose resulting from filtered air intake pressurization to the control room. The NRC performed parametric evaluations and determined that while filtered air

intake pressurization may result in a small addition to the control room whole body dose from noble gases, the increase is more than offset by the reduction in thyroid dose and TEDE from inhalation of radioactive particulates, such as iodine.

Based upon its analyses, the NRC does not agree with the petitioner's assertion regarding the negative safety impact of providing filtered intake flow into the control room. The NRC's performance-based criterion in GDC 19 requires that an applicant provide a control room habitability design that meets the specified dose criterion. Although NRC regulatory guidance provides examples of acceptable design approaches, the approach used to meet the criterion is largely under the control of an applicant. In order to meet this requirement, many licensees have chosen to incorporate filtered air intake pressurization into their control room emergency ventilation designs to reduce the cumulative dose to operators during a DBA. The purpose of providing filtered air intake pressurization flow is to establish positive pressure in the control room relative to the adjacent areas, thereby reducing the quantity of unfiltered air inleakage. Limiting unfiltered inleakage significantly reduces the thyroid dose from inhalation.

2. The petitioner also stated that the current regulation is inconsistent with the goal of allowing operators to remain in the control room in order to mitigate accident consequences. He stated that common designs, such as a filtered air intake pressurization system, which focus on compliance with existing criteria, increase the probability that the control room will have to be evacuated.

The 5 rem control room design criterion is not a maximum integrated dose above which control room evacuation is mandated during an accident. Rather, the criterion provides a design basis to ensure that the control room will maintain a habitable environment for operators to control the plant during a DBA.

The petitioner based his assertion on the assumption that filterable activity is not likely to be a significant contributor to dose in a reactor accident. As an example, the petitioner used the March 1979 Three Mile Island Unit 2 accident. Since the accident, the NRC has expended considerable resources to better define the expected quantity and distribution of activity that could be released during a major reactor accident. As a result of this research, the NRC promulgated 10 CFR 50.67 on December 23, 1999 (64 FR 72001). Under 10 CFR 50.67, a licensee can apply for a license amendment to adopt an alternative source term (AST) that reflects a more realistic assessment of the timing of the release and the quantity and distribution of activity that could be released during a major accident hypothesized for purposes of design analyses. Many licensees have used this approach to comply with NRC regulations governing control room dose.

In addition, 10 CFR 50.67 revised the control room dose criterion from a 5 rem whole body dose, or its equivalent to any organ, to a 5 rem TEDE. The relatively low thyroid organ weighting factor, as defined in 10 CFR 20.1003, ''Definitions,'' and used in the calculation of TEDE, allows for a significant reduction in the controlling aspects of the thyroid dose, which normally governed compliance with control room dose guidelines. The NRC has significantly improved the accuracy of the source term and dose methodology used in design-basis dose consequence analyses. The updated source term and dose methodology address the petitioner's concerns regarding the emphasis on thyroid dose in control room habitability analyses.

3. The petitioner noted that the dose from increased iodine concentration can be mitigated by use of potassium iodide (KI) or respiratory protection, but the current regulations do not permit these mitigation measures to be used in design analyses.

The NRC agrees that KI or Self-Contained Breathing Apparatuses (SCBAs) do have merit as short-term compensatory measures. However, the potential medical complications of KI and the potential adverse impacts to human performance of SCBAs make these measures unsuitable for long-term use. Further, the NRC's policy of ensuring that process or other engineering controls are in place instead of relying on the use of personal protective equipment is clearly set forth in 10 CFR 20.1701, ''Use of process or other engineering controls'' and 10 CFR 20.1702, ''Use of other controls.'' This policy is consistent with the recommendations of international and national radiation protection committees as described in Paragraph 167 of the International Commission on Radiological Protection (ICRP) Publication 26.

Paragraph 167 of ICRP Publication 26 recommends that ''[a]s far as is reasonably practicable, the arrangements for restricting occupational exposure should be applied to the source of radiation and to features of the workplace. The use of personal protective equipment should in general be supplementary to these more fundamental provisions. The emphasis should thus be on intrinsic safety in the workplace and only secondarily on protection that depends on the worker's own actions,'' such as the ingestion of KI or use of respiratory equipment. Further, the use of respiratory equipment by control room personnel during an emergency condition would impede the performance of functions necessary for the protection of public health and safety. Therefore, the NRC has not permitted licensees to rely on either KI or respiratory protection as a permanent solution to demonstrate compliance with the control room radiological dose guidelines, although such measures are available if the fundamental dose design provisions are less effective than anticipated.

4. The petitioner stated that it is inconsistent to provide credit for respiratory protection in control room habitability toxic gas release evaluations, but not for design analyses.

The NRC does not agree with the petitioner. In the case of toxic gas releases, continued plant operation or a normal plant shutdown would be required. In the case of a major reactor accident involving radiological releases, control room personnel must implement extensive emergency operation procedures to ensure public health and safety. Wearing respiratory protection during normal operations or even during an orderly shutdown, should it be necessary as a result of a toxic gas release, would not be expected to present significant challenges to control room personnel equivalent to those present during a reactor accident. The NRC is reluctant to place any more of a burden than is absolutely necessary on control room personnel, who would already be significantly tasked ensuring that all emergency procedures are carried out without error.

B. Procedures Are Not Optimized

The petitioner stated that control room dose mitigation procedures must be consistent with the licensing basis and may not be the optimum mitigation strategy for more likely conditions. For example, he stated that control room dose models do not model dispersion as a period during the day with higher concentrations while the plume is blowing towards the control room and then a period of zero concentration for the rest of the day. Instead, analysis methods simplify this effect by assuming that a lower concentration is present continuously. The petitioner claimed that if procedures were revised to include a control room purge mode

strategy, a ''calculated increase in consequences in the simplistic design basis analysis'' would result.

The NRC disagrees with the petitioner. The NRC's regulations do not require that procedures be limited to the most limiting licensing-basis assumptions. Further, the NRC expects licensees to develop procedures that address the full-scope review of design-basis events and conditions.

With respect to the petitioner's example, procedures to operate the control room in its design-basis mode must be provided. These procedures do not preclude licensees from creating additional procedures to purge the control room if warranted by plant conditions. Licensees are permitted to develop and implement such procedures under existing NRC regulations.

The NRC agrees that control room purging may be a reasonable action during a reactor accident when the level of outside airborne concentration of radioactive material is less than the level inside the control room. However, the conditions favorable for control room purging cannot be predicted, and the NRC cannot credit control room purging in the DBA analysis unless the timing of the release can be accurately established. For accidents where NRC regulatory guidance has established the release duration, the NRC has accepted credit for control room purging after the release has ended. As a design criterion, GDC 19 does not supplant the radiation protection standards of 10 CFR Part 20, which treat the radiation exposure of control room operators as occupational exposure. Therefore, the NRC expects licensees to maintain the accumulated dose of their radiation workers ALARA. During an accident, health physics personnel would monitor the radiological conditions in the control room and other emergency response facilities. These health physicists are responsible for making appropriate recommendations to plant personnel on actions that can be taken to maintain the dose to emergency responders ALARA.

C. Challenges to Safety Systems

The petitioner stated that the current design requirements, which are usually imposed to ensure the assumptions of the dose analysis are met, may not be optimum from an overall risk perspective. As an example, he stated that a common design requirement specifies that the normal control room ventilation must isolate on receipt of a safety injection or containment isolation signal during an assumed loss-of-coolant accident. The petitioner stated that it is more logical to delay control room isolation until radioactivity is detected in the control room or it is known that a radioactive plume is blowing towards the control room. The petitioner suggested that mitigating design strategies should be based on overall risk reduction designed for more likely conditions, not on one unlikely set of fixed hypothetical conditions.

The NRC does not agree with the petitioner. Contrary to the petitioner's assertion, the NRC's regulations do not require immediate control room isolation or immediate appearance at the control room intake of the radioactive plume assumed in design-basis dose consequence analyses. The NRC has approved, in accordance with its regulations, plant designs that do not immediately isolate the control room ventilation system. Further, design bases that include the immediate startup of control room ventilation systems and loading of electrical buses and diesel generators with this equipment do not require operation of plant systems beyond their design capabilities; the diesels are specifically designed and sized to accommodate these safety loads. Therefore, the performance of these systems should not be impacted, and there is no increased risk to public health and safety.

D. Inappropriate Technical Specification Action Statements

The petitioner stated that the conservative nature of the current radiological dose mitigation analyses also results in inappropriate TS action statements. He stated that ''there is insignificant safety significance to the TS associated with control room habitability and yet there are shutdown requirements.'' The petitioner believes that in order to evaluate the net public safety risk associated with these TS shutdown requirements, small but quantifiable public risks associated with the shutdown of a nuclear power plant must be considered, including but not limited to the following:

1. Risk associated with bringing the plant through a transient and another thermal cycle;

2. Airborne pollutants released by the fossil units required to operate to make up for lost power; and

3. Potential for challenging electric power grid stability with the public risk associated with the possibility of rolling blackouts or brownouts or, under the worst conditions of grid instability, the potential for a loss of offsite power at multiple nuclear power facilities.

The petitioner claimed that the shutdown requirement increases the net public risk and should be eliminated because it is only imposed as a ''matter of compliance.''

The NRC disagrees with the petitioner. The NRC has approved license amendments to replace TS requirements for an immediate shutdown for an inoperable control room envelope boundary with requirements for immediate mitigating actions and restoration of the control room envelope to operable status within 90 days.

The NRC has determined that none of the regulations proposed to be changed by the petitioner directly require a plant shutdown in response to control room habitability issues. Existing NRC regulations permit a licensee to propose alternative TS action requirements to its plant shutdown requirements. The NRC notes that even if the petitioner's proposed regulatory changes were made, licensees would still need to submit a license amendment to justify changes to their TSs for NRC approval.

A controlled shutdown and cooldown of a plant is a safe evolution within the design capability of the plant and would not result in undue risk to public safety. In the event of unusual circumstances associated with adverse electrical power grid instability or other complicating issues that would be associated with a plant shutdown, there are processes available for a licensee to obtain regulatory relief to safely continue plant operation (*e.g.*, emergency/exigent technical specification change, enforcement discretion).

E. Unjustified Technical Specification Surveillances

The petitioner stated that ''individual input assumptions for radiological dose analyses have no significance in predicting reality or the acceptability of results. Even if actual conditions were such that one of the assumptions was non-conservative by a couple orders of magnitude, the ultimate result (in this case habitability of the control room) would still be acceptable due to the significant conservatisms in the other assumptions and the simplicity of effective mitigating actions such as the use of KI.'' He stated that although most control room habitability surveillances can be performed with minimal resources, licensees have been required to demonstrate the accuracy of the assumption regarding unfiltered inleakage using an unjustified tracer gas testing method that costs approximately $100,000 per test. The petitioner stated these tests have demonstrated that although inleakage values assumed in the analyses were nonconservative, there was no safety significance and continued operation was justified. The

petitioner concluded that the expenditure for tracer gas testing could be better used for improvements that would likely be more beneficial to plant safety; therefore, the required performance of this test could have a net negative safety consequence. The petitioner stated that previous surveillances, such as a pressurization test, combined with lessons learned from tracer gas testing result in an effective preventative maintenance program.

The NRC does not agree with the petitioner's assertion that individual input assumptions for radiological dose analyses have no significance in predicting reality or the acceptability of results. The NRC places a high priority on operator safety; the requirements contained in GDC 19 should be retained because they provide physical and psychological protection for operators and ultimately for the general public. Therefore, the data used in the analyses to determine operator safety should be accurate, and when data are uncertain, appropriate conservatisms are applied.

The NRC does not agree with the petitioner's statement that the expenditure for tracer gas testing could be better used for improvements that would likely be more beneficial to plant safety nor does the NRC agree that the performance of tracer gas testing could have a net negative safety consequence. The potential dose to the operator must be quantified in order to ensure that the requirements of GDC 19 are met; the specific measurement of inleakage is one of the inputs to the analyses used to quantify the potential dose to the operator. Prior to the use of tracer gas to measure inleakage, the quantity of inleakage was assumed rather than measured and subsequently found to be nonconservative. Tracer gas testing is justified because it ensures operator safety. Other methods of measuring inleakage have not been successfully demonstrated.

F. Petitioner's Proposed Alternatives to Current NRC Guidance

The NRC has decided to deny this petition for rulemaking and would normally not discuss the petitioner's proposed guidance in this document. However, in order to clarify the NRC's decision to maintain the current radiological dose requirements, the following discussion is provided.

Under Commission policy, the NRC's regulations for control room habitability provide performance-based requirements to ensure that plant personnel are adequately protected. The NRC has concluded that prescriptive requirements or guidance, such as that proposed by the petitioner, may unnecessarily restrict a licensee's options for complying with the NRC's regulations.

The petitioner proposed revisions to the NRC's regulatory guidance to help implement his proposed rule change. NRC regulatory guidance is not an appropriate subject for a PRM and the NRC will not generally consider such requests through this process. Further, current NRC regulatory guidance provides one acceptable mechanism for licensees and applicants to meet the requirements of the NRC's regulations. Applicants and licensees may propose alternative means of complying with the NRC's regulations, which will be evaluated by the NRC staff on a case-by-case basis.

1. The petitioner recommended that the control room ventilation system should isolate on the detection of high radiation or toxic intake. The NRC disagrees with the petitioner. All control rooms are required by TSs to take appropriate action upon detection of radiation or toxic gas. Appropriate action may differ from plant to plant depending on location, design, and TSs. Because plants are unique, licensees can demonstrate compliance with the control room design criteria by taking different approaches. The petitioner's suggestion does not address the long-term release situations that would be expected under a worst case accident scenario. Control room isolation alone would not be an acceptable solution because it does not adequately consider the long term breathing air requirements necessary to provide a safe working environment in the control room. After a relatively short period of time, an intake of air into the control room would be necessary. Licensees include these considerations in their site-specific control room habitability analyses. Therefore, the NRC concludes that changing guidance to recommend control room isolation on detection of high radiation or toxic gas is an unnecessarily prescriptive recommendation in comparison to the existing performance-based dose criterion.

2. The petitioner recommended that the control room have a minimum of one foot of concrete shielding (or equivalent) on all surfaces. The NRC disagrees with the petitioner. The NRC believes that control rooms are adequately protected from the effects of direct radiation because current regulations require that either a 5 rem whole body or a 5 rem TEDE acceptance criterion be met under DBA conditions. Licensees include the effects of direct radiation from all potential sources in their control room dose consequence analyses. Typically these sources include the following:

- Contamination of the control room atmosphere by the intake and infiltration of the radioactive material contained in the radioactive plume released from the facility;
- Direct shine from the external radioactive plume released from the facility with credit for control room structural shielding;
- Direct shine from radioactive material in the containment with credit for both the containment and control room structural shielding; and
- Radiation shine from radioactive material in systems and components inside or external to the control room envelope, including radioactive material buildup on the control room ventilation filters.

Many control rooms already have one foot or more of concrete shielding on all surfaces. One foot of concrete shielding does not guarantee adequate protection from radiation. For example, surfaces with 1 foot of concrete with penetrations for various equipment, such as electrical wiring and ventilation ducts, may not provide any more protection than non-concrete surfaces or surfaces with less than 1 foot of concrete. To show compliance with the current control room dose criterion, licensees provide detailed radiological calculations to ensure that under DBA conditions control room personnel will be adequately protected. Licensees have demonstrated compliance with the regulations crediting many different design approaches. The NRC concludes that recommending that the control rooms have one foot of concrete shielding is an unnecessarily prescriptive recommendation.

3. The petitioner recommended that because of the low risk significance of being outside the control room habitability program guidelines, a plant shutdown should not be required in this condition. Rather, the petitioner recommended that the program could specify that timely actions should be taken to return the plant to within the guidelines. If not complete within 30 days, the petitioner suggested that a special report would be sent to the NRC with a justification for continued operations and a proposed schedule for meeting the guidelines. The NRC disagrees with the petitioner that a regulatory change is required to permit these changes to plant TSs. The NRC allows deviations from the integrity of the control room envelope without requiring an immediate plant shutdown.

4. The petitioner recommended that as an alternative to the total removal of

dose guidelines from the regulations, most of his concerns could be resolved if the dose criteria were based solely on the whole body dose from noble gases. The NRC does not agree with the proposition that the dose criteria should be based solely on the whole body dose from noble gases. The control room dose criterion of 5 rem whole body or its equivalent to any organ imposes two requirements on licensees: Satisfaction of the whole body dose criterion, which is generally dominated by the dose from noble gases; and satisfaction of the organ-specific dose guidelines, which are generally dominated by the thyroid dose from the inhalation of iodine. In most cases, demonstrating compliance with thyroid dose guidelines poses a significantly greater challenge to licensees than does compliance with the whole body dose criterion.

The 1999 amendment to 10 CFR 50.67 (64 FR 12117), revised the control room dose limit to allow licensees to show compliance with either the existing limits, using the traditional Technical Information Document (TID)–14844 source term assumptions, or a revised single control room dose criterion of 5 rem TEDE,[1] if the licensee adopts the AST. With the ability to reassess a maximum credible radiological release using the AST, many licensees have shown compliance with the § 50.67 single control room dose criterion of 5 rem TEDE. Licensees have accomplished this while achieving an enhanced degree of operational flexibility not realized using the traditional TID–14844 source term with the associated whole body dose criterion and organ dose guidelines. Because compliance with § 50.67 is demonstrated by calculating the TEDE, the relative contribution of the thyroid dose to the demonstration of compliance with the control room criterion has been substantially and appropriately reduced. In addition, many licensees that continue to use the traditional TID–14844 source term have incorporated the guidance in Regulatory Guide (RG) 1.195, ''Methods and Assumptions for Evaluating Radiological Consequences for Design-Basis Accidents at Light-Water Nuclear Power Reactors'' (ML031490640) to achieve operational flexibility. Following the guidance in RG 1.195, licensees are able to evaluate control room habitability using a 50 rem thyroid dose guideline. This represents a significant relaxation from the 30 rem thyroid dose guideline that was incorporated into previous guidance documents.

The petitioner also stated that the whole body dose from noble gases is likely to be the only possible dose impact that may result in control room evacuation. The NRC does not accept the premise that any maximum credible radiological release would result in the necessity for a control room evacuation. As stated previously, the 5 rem control room design criterion is not intended to be a maximum integrated dose level at which control room evacuation would be mandated during an accident. Rather, the criterion is used as a design basis to ensure that the control room, by design, will provide a habitable environment for the control of the plant under the maximum credible radiological release conditions, and as such will provide reasonable assurance of adequate protection.

The petitioner stated that most of his concerns would be resolved if credit for SCBAs or KI was allowed in the analysis of the dose from iodines and particulates. The NRC does not agree with the option of replacing engineering controls for radiological protection with credit for personal protective equipment. As discussed previously, the option of allowing credit for SCBAs or KI to show compliance with the control room performance-based design criterion is inimical to the NRC design philosophy incorporated into 10 CFR Part 20, as well as international standards for radiological protection as set forth in ICRP Publication 26.

IV. Public Comments

1. Overview of Public Comments

The NRC's notice of receipt and request for public comment invited interested persons to submit comments. The comment period for PRM–50–87 closed on September 25, 2007. The NRC reviewed and considered the comments in its decision to deny the petition. The NRC received two public comments, one from Mr. Walston Chubb (ML072681072), and one from Mr. James H. Riley on behalf of the Nuclear Energy Institute (NEI) (ML072690232).

2. Mr. Walston Chubb Comment

Comment: Mr. Chubb recommended that operators be required to remain on duty until they are relieved or their short-time doses are between 100 and 200 rem.

NRC Response: The primary objective of GDC 19 is to ensure that the design of the control room and its habitability systems provide a ''shirt-sleeved'' environment for operators during both normal and accident conditions. This environment facilitates operator response to normal and accident conditions while minimizing errors of omission or commission. Another objective is to ensure that the radiation dose levels in the control room would make it the safest location on site, thereby allowing the operators to remain in the control room. Any reduction in operator accident response capabilities may negatively impact public health and safety.

The NRC's decision to apply the 5 rem whole body dose criterion was based on the following:

- A whole body radiation exposure of 5 rem is considered unlikely to cause increased anxiety that would result in operator impairment, since the criterion is comparable to the occupational dose limits.
- A whole body radiation exposure of 5 rem would not result in any somatic response that could result in operator impairment. Generally, the onset of clinically observable somatic effects occurs between 25 and 50 rem.
- GDC 19, as a design criterion, does not supplant the radiation protection standards of 10 CFR Part 20. The radiation exposure of control room operators is controlled, as for any radiation worker at the facility, as occupational exposure under 10 CFR Part 20. In the statements of consideration for the 10 CFR Part 20 rulemaking (56 FR 23365; May 21, 1991), the NRC stated that the dose limits for normal operation should remain the primary guidelines for an emergency.

The statement of considerations in the proposed and final rule amending 10 CFR 50.67 and GDC 19 (64 FR 12117, March 31, 1999; and 64 FR 71990, December 23, 1999, respectively) included the NRC's basis for establishing the 5 rem TEDE as the GDC 19 numeric criterion for licensees applying for amendment under 10 CFR 50.67. It also reaffirmed the position that the criteria in GDC 19 and the final rule are based on occupational exposure limits.

The 5 rem control room design criterion is not intended to be a maximum integrated dose above which control room evacuation would be mandated during an accident. Rather, the 5 rem design criterion ensures that the control room, by design, will provide a habitable environment for the

[1] As defined in 10 CFR 20.1003, ''Total Effective Dose Equivalent (TEDE) means the sum of the effective dose equivalent (for external exposures) and the committed effective dose equivalent (for internal exposures).'' The effective dose equivalent for external exposures includes the whole body dose from noble gases. The committed effective dose equivalent for internal exposure includes the thyroid dose from inhalation of iodine.

control of the plant under all DBA conditions.

Providing a safe working environment for the highly skilled professionals needed to operate a nuclear power plant is a primary objective of NRC regulations related to occupational and accident dose, and it is a paramount goal throughout the entire nuclear power industry. The NRC concludes that the proposal to set the control room design criterion at 100 rem, which is well above the level at which the onset of clinically observable somatic effects would occur, is antithetical to the fundamental principle of protecting public health and safety and is not acceptable.

3. NEI Comments

NEI provided the following comments:

Comment: ''It is not so much the value of the exposure limits that is the problem. The NRC should be more open to other methods of analysis proposed by licensees. Every Regulatory Guide states that the guidance is one method acceptable to the staff and that other methods proposed by licensees will be evaluated on a case-by-case basis. However, in practice it is often difficult to justify different approaches.''

NRC Response: To the extent that the comment implicitly criticizes the NRC for allegedly failing to consider alternatives for compliance with GDC 19 and 10 CFR 50.67 in a manner other than that suggested in a regulatory guide, that concern is beyond the scope of this petition for rulemaking. Further, the commenter presented no basis for this implicit criticism—the NRC routinely considers licensee and applicant-proposed alternatives to methods set forth in a Regulatory Guide. However, the NRC expects licensees and applicants to provide technically sufficient basis for the use of an alternative for compliance with an NRC regulation, which is also consistent with the regulatory policies of the NRC. That a licensee or applicant may find it difficult to provide sufficient basis justifying the use of an alternative approach, however, would not appear to present a valid regulatory concern.

Comment: Existing emergency filtration systems should be maintained to practical performance criteria. NEI stated that this area has a lot of potential for improvement and gave the following examples:

- The current practice (*i.e.*, RG 1.52, ''Design, Inspection, and Testing Criteria for Air Filtration and Adsorption Units of Post-Accident Engineered-Safety-Feature Atmosphere Cleanup Systems in Light-Water-Cooled Nuclear Power Plants'') (ML011710176) is to apply a safety factor of 2 for laboratory testing of charcoal beds. The actual efficiencies are typically much higher than those allowed by RGs.
- Some plants have an 8-inch charcoal bed, for which only 4 inches is allowed to be credited.
- Other plants have filtration systems in series, for which only one composite filter can be credited.

NRC Response: The NRC's position on existing emergency filtration systems is outlined in RG 1.52, Revision 3, issued June 2001. The previous revision of the RG included a safety factor as great as 7 whereas Revision 3 includes a safety factor of 2 to account for degradation of the system between test periods. A safety factor represents margin in the capability of the adsorbent (carbon) installed in the system to perform the required safety function. Because carbon can degrade between test periods, a safety factor provides confidence that the anticipated degradation will not be beyond the minimum level necessary to perform its required safety function.

RG 1.52, Revision 3, indicates that a 4-inch carbon bed in U.S. nuclear power plants is 99 percent efficient, with a safety factor of 2 and a penetration (as defined in American Society for Testing and Materials D 3803–89) of less than or equal to 0.5 percent. The NRC believes that a 4-inch carbon bed thickness is sufficient to provide adequate protection, and that the 4 inches, as reflected in the RG, is not intended to be an upper limit on bed thickness. It is acceptable to provide additional carbon that may include 6 inches, 8 inches, or even greater bed thickness. The NRC also believes there are benefits provided by carbon bed thicknesses greater than 4 inches that are not reflected in the RG. The benefits may include longer bed life contributing to lower overall cost.

With respect to filtration systems in series, they are treated as a composite (*i.e.*, the sum of individual filters in series). For example, the efficiency of two 2-inch beds in series is the same as one 4-inch bed.

Comment: In response to the petitioner's statement that current TS for system performance should be eliminated and that the administrative portion of the TS could include a requirement to have a control room habitability program, NEI commented, ''This recommendation is covered by TSTF–448 and GL 2003–01.''

Response: NRC agrees with the comment. NRC prepared and made available a model safety evaluation (SE) and a model no-significant-hazards-consideration (NSHC) determination relating to the modification of technical specification (TS) requirements regarding the habitability of the control room envelope (CRE) for referencing in license amendment requests (LARs). NRC also made available an associated model LAR for use by licensees to prepare such LARs. The TS modification is based on NRC staff approved changes to the improved standard technical specifications (STS) (NUREGs 1430–1434, available on NRC's public Web site at *www.nrc.gov/reactors/operating/licensing/techspec/current-approved-sts.html*) that were proposed by the pressurized and boiling water reactor owners groups' Technical Specifications Task Force (TSTF) on behalf of the commercial nuclear electrical power generation industry, in STS change traveler TSTF–448, Revision 3 (ML063460558). NRC published a Notice of Availability of the SER in the **Federal Register** on January 17, 2007 (72 FR 2022). Generic Letter (GL) 2003–01, dated June 12, 2003, is available on ADAMS (ML031620248).

Comment: In response to the petitioner's proposed guidance, NEI provided the following comments:

- The control room ventilation system should isolate on the detection of high radiation or toxic gas intake. NEI commented, ''A good many control rooms in the industry already operate in this manner. Conversely, there are some plants that do not have automatic initiation of the emergency mode. Making this a requirement could result in an undue (and expensive) modification/backfit. For those plants susceptible to toxic gas intrusion, automatic initiation is typically the case (although not specifically implemented in all cases). If required, this also could result in undue (and expensive) modifications.''
- The control room should have a minimum of one foot of concrete shielding (or equivalent) on all surfaces. NEI commented, ''It is unlikely that all control rooms have one foot of concrete shielding on all surfaces. This requirement could result in undue (and expensive) modifications. A similar concern applies to the technical support center, which may also be affected by this requirement.''
- SCBAs and KI tablets should be readily available for operator use. Operators should maintain training in SCBAs. NEI commented, ''The use of these methods has merit, but additional evaluation of their effects is necessary. The medical complications of ingesting KI would have to be evaluated for all CR personnel. The use of SCBA credit would require specific training for which operators will need to demonstrate the ability to conduct their

safety-related functions while wearing a SCBA for several hours.''

• Procedures should be developed to ensure control room purging is considered when the outside concentration is less than the inside concentration. NEI commented, ''Although this appears to be a good practice, it can't be credited in the operator dose analysis. The timing of purging could be critical based on the timing of the release and the release pathway. Therefore, this recommendation may not have any practical merit.''

The petitioner stated that because of the low risk significance of being outside the control room habitability program guidelines, a plant shutdown would not be required in this condition; rather, the program could specify that timely actions should be taken to return the plant within the guidelines. If not complete within 30 days, a special report would be sent to the NRC with a justification for continued operation and a proposed schedule for meeting the guidelines. NEI commented, ''This is a valid point that the industry supports.''

The petitioner stated that as an alternative to total removal of dose guidelines from the regulations, most of his concerns could be resolved if the dose criteria were based solely on the whole body dose from noble gases that he believes is the only possible dose impact that may result in control room evacuation. NEI commented, ''It is not clear that the noble gas contribution would be limiting in all cases. However, this may be the case if KI were allowed to be credited.''

Response: These comments have been addressed in Section III of this document.

V. Denial of Petition

Based upon review of the petition and comments received, the NRC has determined that the conclusions upon which the petitioner relies do not substantiate a basis to eliminate the control room radiological dose acceptance criteria from current regulations as requested. For the reasons discussed previously, the Commission denies PRM–50–87.

Dated at Rockville, Maryland, this 14th day of January 2009.

For the Nuclear Regulatory Commission.

Annette L. Vietti-Cook,

Secretary of the Commission.

[FR Doc. E9–1211 Filed 1–23–09; 8:45 am]

BILLING CODE 7590–01–P

DEPARTMENT OF TRANSPORTATION

Federal Aviation Administration

14 CFR Part 25

[Docket No. NM398; Notice No. 25–09–01–SC]

Special Conditions: Model C–27J Airplane; Interaction of Systems and Structures

AGENCY: Federal Aviation Administration (FAA), DOT.

ACTION: Notice of proposed special conditions.

SUMMARY: This action proposes special conditions for the Alenia Model C–27J airplane. This airplane has novel or unusual design features when compared to the state of technology described in the airworthiness standards for transport-category airplanes. These design features include electronic flight-control systems. These special conditions pertain to the effects of novel or unusual design features such as effects on the structural performance of the airplane. We have issued additional special conditions for other novel or unusual design features of the C–27J.

The applicable airworthiness regulations do not contain adequate or appropriate safety standards for this design feature. These proposed special conditions contain the additional safety standards that the Administrator considers necessary to establish a level of safety equivalent to that established by the existing airworthiness standards.

DATES: We must receive your comments by February 25, 2009.

ADDRESSES: You must mail two copies of your comments to: Federal Aviation Administration, Transport Airplane Directorate, Attn: Rules Docket (ANM–113), Docket No. NM398, 1601 Lind Avenue SW., Renton, Washington 98057–3356. You may deliver two copies to the Transport Airplane Directorate at the above address. You must mark your comments: Docket No. NM398. You can inspect comments in the Rules Docket weekdays, except Federal holidays, between 7:30 a.m. and 4 p.m.

FOR FURTHER INFORMATION CONTACT: Holly Thorson, FAA, International Branch, ANM–116, Transport Airplane Directorate, Aircraft Certification Service, 1601 Lind Avenue SW., Renton, Washington 98057–3356; telephone (425) 227–1357, facsimile (425) 227–1149.

SUPPLEMENTARY INFORMATION:

Comments Invited

We invite interested people to take part in this rulemaking by sending written comments, data, or views. The most helpful comments reference a specific portion of the special conditions, explain the reason for any recommended change, and include supporting data. We ask that you send us two copies of written comments.

We will file in the docket all comments we receive, as well as a report summarizing each substantive public contact with FAA personnel concerning these special conditions. You can inspect the docket before and after the comment closing date. If you wish to review the docket in person, go to the address in the **ADDRESSES** section of this preamble between 7:30 a.m. and 4 p.m., Monday through Friday, except Federal holidays.

We will consider all comments we receive on or before the closing date for comments. We will consider comments filed late if it is possible to do so without incurring expense or delay. We may change these special conditions based on the comments we receive.

If you want the FAA to acknowledge receipt of your comments on this proposal, include with your comments a self-addressed, stamped postcard on which the docket number appears. We will stamp the date on the postcard and mail it back to you.

Background

On March 27, 2006, the European Aviation Safety Agency (EASA) forwarded to the FAA an application from Alenia Aeronautica of Torino, Italy, for U.S. type certification of a twin-engine commercial transport designated as the Model C–27J. The C–27J is a twin-turbopropeller, cargo-transport aircraft with a maximum takeoff weight of 30,500 kilograms.

Type Certification Basis

Under the provisions of Section 21.17 of Title 14 Code of Federal Regulations (14 CFR) and the bilateral agreement between the U.S. and Italy, Alenia Aeronautica must show that the C–27J meets the applicable provisions of 14 CFR part 25, as amended by Amendments 25–1 through 25–87. Alenia also elects to comply with Amendment 25–122, effective September 5, 2007, for 14 CFR 25.1317.

If the Administrator finds that existing airworthiness regulations do not adequately or appropriately address safety standards for the C–27J due to a novel or unusual design feature, we prescribe special conditions under provisions of 14 CFR 21.16.

Source: U.S. Environmental Protection Agency

Exposure (rem)	Health Effect [Acute]	Time to Onset (without treatment)
5-10	changes in blood chemistry	
50	nausea	hours
55	fatigue	
70	vomiting	
75	hair loss	2-3 weeks
90	diarrhea	
100	hemorrhage	
400	possible death	within 2 months
1,000	destruction of intestinal lining	
	internal bleeding	
	and death	1-2 weeks
2,000	damage to central nervous system	
	loss of consciousness;	minutes
	and death	hours to days

Source: Centers for Disease Control and Prevention (CDC)

Potential Health Effects (Other Than Cancer) of Prenatal Radiation Exposure

Acute Radiation Dose* to the Embryo/Fetus	Time Post Conception				
			Fetogenesis		
	Blastogenesis (up to 2 wks)	Organogenesis (2 –7 wks)	(8–15 wks)	(16 –25 wks)	(26 –38 wks)
< 0.05 Gy (5 rads)†	**Noncancer health effects NOT detectable**				
0.05–0.50 Gy (5–50 rads)	*Incidence of failure to implant may increase slightly, but surviving embryos will probably have no significant (noncancer) health effects*	*• Incidence of major malformations may increase slightly* *• Growth retardation possible*	*• Growth retardation possible* *• Reduction in IQ possible (up to 15 points, depending on dose)* *• Incidence of severe mental retardation up to 20%, depending on dose*	***Noncancer health effects unlikely***	

Source: Centers for Disease Control and Prevention (CDC)

Potential Health Effects (Other Than Cancer) of Prenatal Radiation Exposure

Acute Radiation Dose* to the Embryo/Fetus	Time Post Conception				
			Fetogenesis		
	Blastogenesis (up to 2 wks)	Organogenesis (2 –7 wks)	(8–15 wks)	(16 –25 wks)	(26 –38 wks)
> 0.50 Gy (50 rads) *The expectant mother may be experiencing acute radiation syndrome in this range, depending on her whole-body dose.*	Incidence of failure to implant will likely be large,‡ depending on dose, but surviving embryos will probably have no significant (noncancer) health effects	• Incidence of miscarriage may increase, depending on dose • Substantial risk of major malformations such as neurological and motor deficiencies • Growth retardation likely	• Incidence of miscarriage probably will increase, depending on dose • Growth retardation likely • Reduction in IQ possible (> 15 points, depending on dose) • Incidence of severe mental retardation > 20%, depending on dose • Incidence of major malformations will probably increase	• Incidence of miscarriage may increase, depending on dose • Growth retardation possible, depending on dose • Reduction in IQ possible, depending on dose • Severe mental retardation possible, depending on dose • Incidence of major malformations may increase	Incidence of miscarriage and neonatal death will probably increase depending on dose§

Source: Centers for Disease Control and Prevention (CDC)

Potential Health Effects (Other Than Cancer) of Prenatal Radiation Exposure

Acute Radiation Dose* to the Embryo/Fetus	Time Post Conception				
			Fetogenesis		
	Blastogenesis (up to 2 wks)	Organogenesis (2 –7 wks)	(8–15 wks)	(16 –25 wks)	(26 –38 wks)
Note: This table is intended only as a guide. The indicated doses and times post conception are approximations.					

Gestational age and radiation dose are important determinants of potential noncancer health effects. The following points are of particular note:

- Before about 2 weeks gestation (i.e., the time after conception), the health effect of concern from an exposure of > 0.1 gray (Gy) or 10 rads1 is the death of the embryo. If the embryo survives, however, radiation-induced noncancer health effects are unlikely, no matter what the radiation dose. Because the embryo is made up of only a few cells, damage to one cell, the progenitor of many other cells, can cause the death of the embryo, and the blastocyst will fail to implant in the uterus. Embryos that survive, however, will exhibit few congenital abnormalities.

- In all stages of gestation, radiation-induced noncancer health effects are not detectable for fetal doses below about 0.05 Gy (5 rads). Most researchers agree that a dose of < 0.05 Gy (5 rads) represents no measurable noncancer risk to the embryo or fetus at any stage of gestation. Research on rodents suggests a small risk may exist for malformations, as well as effects on the central nervous system in the 0.05–0.10 Gy (5–10 rads) range for some stages of gestation. However, a practical threshold for congenital effects in the human embryo or fetus is most likely between 0.10–0.20 Gy (10–20 rads).

- From about 16 weeks' gestation to birth, radiation-induced noncancer health effects are unlikely below about 0.50 Gy (50 rads). Although some researchers suggest that a small possibility exists for impaired brain function above 0.10 Gy (10 rads) in the 16- to 25-week stage of gestation, most researchers agree that after about 16 weeks' gestation, the threshold for congenital effects in the human embryo or fetus is approximately 0.50–0.70 Gy (50–70 rads).

The following table presents the estimates of the fatal cancer risk for a group of 1,000 workers of various ages, assuming that each worker received an acute dose of 25 rems (0.25 Sv) in the course of assisting in an emergency. The estimates show that a 25-rem emergency dose might increase an individual's chances of developing fatal cancer from about 20% to about 21%.

Risk of Premature Death from Exposure to 25 rems (0.25-Sv) Acute Dose

Age at Exposure (years)	Estimated Risk of Premature Death (Deaths per 1,000 Persons Exposed)
20-30	9.1
30-40	7.2
40-50	5.3
50-60	3.5

Source: EPA-400-R-92-001 "Manual of Protective Action Guides and Protective Actions for Nuclear Incidents," May 1992.

Fetal Radiation Dose Estimates

A PERSPECTIVE ON RISK TO THE FETUS FROM IONIZING RADIATION

Ionizing radiation is known to cause harm in mammalian organisms. Deleterious effects of radiation include carcinogenicity, mutagenicity and organ system toxicity. As general rule, the sensitivity of a tissue to radiation is directly proportional to its rate of proliferation. Therefore, one could infer that the human fetus, because of its rapid progression from a single cell to a formed organism in nine months, is more sensitive to radiation than the adult. This inference is supported by the results of experiments in animal models, and experience with human populations that have been exposed to very high doses of radiation (atomic bombing victims). In humans, the major deleterious effects on the fetus include fetal wastage (miscarriage), teratogenicity (birth defects), mental retardation, intrauterine growth retardation and the induction of cancers (such as leukemia) that appear in childhood. Birth defects and mental retardation are the adverse effects which are of the most immediate concern for expectant mothers. Fortunately, not all exposures to ionizing radiation result in these outcomes. The risk to the fetus is a function of (a) gestational age at exposure and (b) the radiation dose.

At the level of most diagnostic procedures (fetal dose < 10 rem), little data in humans is available. However, some qualitative observations regarding fetal risk can be made.

Risk Related to Gestational Age

Early Gestation / First Trimester -- At this point, the rate of fetal growth is very rapid and the fetus, *as an organism*, is at its most radiation-sensitive stage if fetal demise is taken as an end-point. The incidence of fetal wastage consequential to radiation exposure at this stage of gestation is not known, since (a) many women were never aware they were pregnant at the time of the exposure or miscarriage, and (b) the "background" rate of miscarriage is believed to be high (25 - 50 percent of conceptions). It is believed that radiation injury during early gestation is an "all-or-nothing" effect.

Second Trimester -- During this period, the overall growth rate of the fetus has slowed. However, the major organ systems are beginning to differentiate. From a standpoint of *future development*, the fetus is in its most sensitive stage. The incidence of gross congenital malformations and mental retardation are dose-related and appear to have thresholds; i.e. doses below which the incidence above "background" is not elevated.

Third Trimester -- Irradiation during this period may deplete cell populations at very high doses (over 50 rem), but will not result in gross organ malformations.

Risk Related to Radiation Dose

The risk of deleterious effects increases with increasing dose. The nature of this dependence, i.e. the shapes of the dose-response curves for humans in the low-dose range (under 50 rem), is controversial. For some prenatal irradiation effects, there is epidemiological basis for the existence of threshold doses. For others, such as childhood cancer induction, the existence of a threshold is not clear-cut. Despite these uncertainties in the dose-effect relationship, some broad generalizations based on fetal dose ranges may be made.

Fetal Dose Less Than 1,000 millirem -- There is no evidence supporting the increased incidence of any deleterious developmental effects on the fetus at diagnostic doses within this range.

Fetal Dose between 1,000 millirem and 10,000 millirem -- The additional risk of gross congenital malformations, mental retardation, intrauterine growth retardation and childhood cancer is believed to be low compared to the baseline risk. However, the lower limits (in terms of statistical confidence intervals around the mean) for threshold doses for some studies, especially those related to cancer induction, fall within this range.

Fetal Dose Exceeding 10,000 millirem -- The lower limits (in terms of statistical confidence intervals) for threshold doses for effects such as mental retardation and diminished IQ and school performance fall within this range. Overall, exposure at levels exceeding 10 rem could be expected to result in a dose-related increased risk for deleterious effects. For example, the lower limit (95% confidence interval) for the threshold for mental retardation is about 15 rem, which an expectation value of about 30 rem.

A Brief History of Radiation

Health physics is concerned with protecting people from the harmful effects of ionizing radiation while allowing its beneficial use in medicine, science, and industry. Since the discovery of radiation and radioactivity 100 years ago, radiation protection standards and the philosophy governing those standards have evolved in somewhat discrete intervals. The changes have been driven by two factors—new information on the effects of radiation on biological systems and changing attitudes toward acceptable risk. The earliest limits were based on preventing the onset of such obvious effects as skin ulcerations that appeared after intense exposure to radiation fields. Later limits were based on preventing delayed effects such as cancer that had been observed in populations of people receiving high doses, particularly from medical exposures and from the atomic-bomb exposures in Hiroshima and Nagasaki.

During the evolution of standards, the general approach has been to rely on risk estimates that have little chance of underestimating the consequences of radiation exposure. It is important to realize that most of the effects observed in human populations have occurred at high doses and high dose rates. The information gathered from those populations must be scaled down to low doses and low dose rates to estimate the risks that occur in occupational settings.

Immediately after the discoveries of x rays in 1895 and radioactivity in 1896, x-ray devices and radioactive materials were applied in physics, chemistry, and medicine. In the very early days, the users of x rays were unaware that large radiation doses could cause serious biological effects. They also had no instruments to measure the strength of the radiation fields. Instead, the calibration of x-ray tubes was based on the amount of skin reddening (erythema) produced when the operator placed a hand directly in the x-ray beam. The doses needed to produce erythema are very high indeed—if the skin is exposed to 200-kilovolt x rays at a high dose rate of 30 rad per minute, then erythema appears after about 20 minutes (or 600 rad) of exposure, and moist desquamation (equivalent to a third-degree burn) occurs after about 110 minutes (or about 2000 rad) of exposure.

Early ignorance of the hazards of radiation resulted in numerous unexpected injuries to patients, physicians, and scientists, and as a result, some researchers took steps to publicize the hazards and set limits on exposure. In July 1896, only one month after the discovery of x rays, a severe case of x-ray-induced dermatitis was published, and in 1902, the first dose limit of about 10 rad per day (or 3000 rad per year), was recommended. The 10 rad-per-day limit was based not on biological data but rather on the lowest amount that could be easily detected, namely, the amount required to produce an observable exposure, or fogging, on a photographic plate. By 1903, animal studies had shown that x rays could produce cancer and kill living tissue and that the organs most vulnerable to radiation damage were the skin, the blood-forming organs, and the reproductive organs.

In September 1924 at a meeting of the American Roentgen Ray Society, Arthur Mutscheller was the first person to recommend a “tolerance” dose rate for radiation workers, a dose rate that in his judgement could be tolerated indefinitely. He based his recommendation on observations of physicians and technicians who worked in shielded work areas. He estimated that the workers had received about one-tenth of an erythema dose per month (or about 60 rem per month) as measured by the x-ray-tube current and voltage, the filtration of the beam, the distance of the workers from the x-ray tube, and the exposure time. He also observed that none of the individuals had shown any signs of radiation injury. He concluded that the dose-rate levels in the shielded rooms were acceptable, but in proposing a tolerance dose, he applied a safety factor of ten and recommended that the tolerance limit be set at one-hundredth of an erythema dose per month (equivalent to about 70 rem per year). A tolerance dose was "assumed to be a radiation dose to

which the body can be subjected without production of harmful effects." Mutscheller presented his recommendation in a paper entitled, "Physical Standards of Protection Against Roentgen Ray Dangers," which was published in 1925. Quite fortuitously, F. M. Sievert arrived at about the same limits using a similar approach.

In 1934, the U.S. Advisory Committee on X-ray and Radium Protection proposed the first formal standard for protecting people from radiation sources. By then the quantitative measurement of ionizing radiation had become standardized in units of roentgens,[1] and therefore, the recommended limit on dose rate was expressed as 0.1 roentgen per day. That value was in line with Mutscheller's recommendation of one-hundredth of an erythema dose per month, and in fact, the two tolerance limits differed only by a factor of two. Whether that difference was due to a rounding factor or a technical difference in the way the roentgen was measured in the U.S. versus Europe is open to interpretation.

It is worth emphasizing that those early limits on exposure to x rays were not arrived at through quantitative observation of biological changes but rather through a judgement call based on the absence of observed biological harm.

The dose limits for radiation sources outside of the body (external sources) were augmented in 1941 by a limit on the amount of radium a person could tolerate inside the body (radium tends to be retained by the body, and because of its long radioactive half-life, it thereby becomes a relatively constant internal source of radiation). The devastating experiences of the radium-dial painters and the origin of the radium standard are described in "Radium—The Benchmark for Internal Alpha Emitters" (see page 224). Decade-long clinical observations of twenty-seven persons who were exposed internally to radium, in combination with quantitative measurements of their radium body burdens, were the basis for the radium standard. In particular, it appeared that the retention of 1.0 microgram or more was required to produce deleterious effects. Applying a safety factor of ten to that result, the committee members responsible for recommending a standard (many of whom had performed the clinical research on the radium patients) suggested that 0.1 microgram (or 0.1 microcurie) of radium would be an appropriate tolerance limit. Again, the ultimate criteria used was a judgement call: They all agreed that they would feel comfortable even if their own children had that amount in their bodies. That initial standard has essentially remained in effect up to the present.

In 1944, the radium standard was used as a basis for setting the first tolerance limit for internal retention of plutonium. A working-lifetime limit of 5 micrograms (0.3 microcuries) was proposed on the basis that plutonium was long-lived and would be a bone-seeker like radium and that the alpha-particle emissions from 5 micrograms of plutonium would deposit ionizing energy at the same rate as the alpha emissions from the allowed 0.1 microgram of radium. In 1945, as a result of animal studies on the relative toxicity of plutonium and radium and on their relative distribution in the body, the Manhattan Engineer District reduced the plutonium limit a factor of 5 to 0.06 microcuries. The Hanford Site, where plutonium was being produced in reactors, reduced the limit even further to 0.03 microcuries. Although today's standards are expressed in terms of an annual inhalation limit rather than a maximum permissible body burden, the current limit recommended by the International Commission on Radiation Protection (ICRP) translates to a body burden that is about the same as the working-lifetime limit set at Hanford during World War II. The concern for limiting and monitoring intakes of radium and plutonium were the beginnings of the field of internal radiation dosimetry.

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[1] The roentgen, the first formal radiation unit, was adopted in 1928 and specifies the quantity of ionizing radiation in terms of the amount of electrostatic charge it produces passing through a volume of air. In particular, the Roentgen is defined as that amount of ionizing radiation that produces 1 electrostatic unit of negative charge in 0.00129 gram of air (1 cubic centimeter of air at standard temperature and pressure). For x rays, 1 rad = 1 rem = 0.96 roentgen.

A great deal of research, particularly animal studies, on the biological effects of radiation were carried out during and immediately after World War II. In 1949 the United States, Canada, and Great Britain held a conference at Chalk River, Ontario, on permissible doses and then published the Tripartite report in which all radiation protection information that had been gathered was discussed and collated. A number of new concepts concerning the measurement of dose had been developed through animal studies. These included *absorbed dose* (measured in rad), *dose-equivalent* (measured in rem), *relative biological effectiveness* (RBE), which relates the rad to the rem for different types of radiations, the absorbed dose as a function of photon energy and depth in tissue (depth dose), the radiotoxicity of plutonium, and the concept of a reference anatomical human. The Tripartite report also recommended standards for internal and external radiation protection, including a plutonium body-burden limit of 0.03 microcuries, a limit on the bone-marrow dose of 300 millirem per week (about 15 rem per year), and a limit on the skin dose of 600 millirem per week (a factor of 2 lower than the value initially recommended by Mutscheller in his 1925 publication). With the exception of the plutonium limit, those values were adopted by the ICRP and the National Council on Radiation Protection and Measurements (NCRP, the new name for the old U.S. Advisory Committee) in 1953 and 1954, respectively. (The plutonium limit recommended by the ICRP was somewhat higher at 0.04 microcuries for the maximum permissible amount of plutonium-239 fixed in the body.)

During the 1950s, further reductions in the standards for external radiation were made as a result of studies on the survivors of the two nuclear weapons dropped on Japan and studies of survivors of high-dose medical procedures. In particular, an early analysis of data from the Japanese atomic-bomb survivors indicated an apparent change in the ratio of the number of males to females among infants born to survivors. At the same time, data from experiments on mammals and fruit flies demonstrated that genetic changes could be induced from very high radiation exposures. Thus, radiation-induced genetic effects became a dominant concern in the early 1950s and led to the first recommended standards for annual dose limits to the public. Later analyses indicated that the early assessment of the atomic-bomb survivors was incorrect, and to this day, radiation-induced genetic changes in humans have never been observed. Nevertheless, the fear of future genetic effects lingered on and probably inspired the creation of such science fiction characters as Godzilla, the Incredible Shrinking Man, Spiderman, the Incredible Hulk, and many others. The concern also led to a reduction in radiation protection standards.

In 1957, the ICRP recommended an annual occupational dose limit of 5 rem per year, and in 1958 the NCRP recommended a life-time occupational dose limit of [(age in years - 18) x 5] rem, or a limit of 235 rem for someone who works from ages 18 to 65. The NCRP also recommended an annual limit to the public of 500 millirem per year. In 1960, the Federal Radiation Council recommended an annual limit of 500 millirem per year for an individual in the general public and a limit of 170 millirem per year as the average annual dose to a population group.

By 1961, it was generally understood that the risk of genetic effects had been overestimated in studies of the atomic-bomb survivors, but another risk was becoming apparent—studies of cancer incidence and mortality among the survivors were beginning to show elevated rates for leukemia. As time passed, elevated rates for solid-tumor cancers were also observed. Those findings as well as other studies led to the understanding that different cancers have different latency periods, or elapsed times, between irradiation of the individual and clinical observation of a malignancy. Solid tumors have latency periods of 25 to 40 years, and leukemia has a latency period of 2 to 25 years. The latency periods generally hold true irrespective of the particular agent that serves as the carcinogen.

The unmistakable appearance of an increased rate of cancer among the atomic-bomb survivors had a

profound impact on the radiation protection community—it brought into focus the possibility that even low levels of exposure might induce cancers. Of course, the data regarding malignancies were obtained from populations receiving high doses at high dose rates. Risks estimates for low doses could only be made by extrapolating the high-dose data, and that procedure suggested that the cancer risks from low doses were small. Nevertheless, there were no data to suggest the existence of a threshold dose for radiogenic cancers, so the small risk per person at low doses had to be considered in relation to the large number of workers who were receiving those doses.

Those considerations resulted in a philosophical shift from mere compliance with dose limits and the avoidance of deterministic effects (such as cataracts and permanent damage to organs) to an emphasis on reducing overall cancer risks to working populations. The ICRP defined a system of dose control consisting of three parts: justification, optimization, and limitation. Justification requires that no new practice involving radiation shall be allowed unless its introduction produces a positive net benefit. Optimization requires that all doses shall be kept *as low as reasonably achievable* (ALARA) taking into account the relevant economic and social factors. Limitation requires that any individual dose not exceed limits set for appropriate circumstances. In today's applications of the dose-control concept, justification and optimization dominate. (More to the point, subjective judgements of regulators rather than the mathematics of optimization often drive the dose limits to lower and lower levels; economic factors are often ignored; and the net result is to make operations involving radiation and radioactive materials extremely expensive.)

In 1977, the ICRP adopted a more formal risk-based approach to setting standards. That approach required that the average incremental risk of death from radiation exposure to workers in radiation industries be no larger than the average incremental risk of death from traumatic injuries to workers in "safe" industries. The incremental risk of death in safe industries is one in ten-thousand, or 10^{-4}, per year. Studies of the atomic-bomb survivors had shown that the risk coefficient for radiation-induced cancer mortality was about 10^{-4} per rem. Based on that risk coefficient, the ICRP recommended a maximum annual dose limit to a radiation worker of 5 rem per year. The 5-rem annual limit was set under the assumption that the average dose would be less than 1 rem per year, and, thus, the average risk of death would be the same as for safe industries. Thus, the new 1977 limit was unchanged from the 1957 limit, but it was now justified in terms of a risk-based philosophy.

During the 1980s, estimates of the doses received by the atomic-bomb survivors were adjusted downward based on new estimates of the ratio of neutrons to gamma rays in the radiation produced by the bomb. Also, new data on cancer incidence and mortality among the survivors indicated higher rates for some cancers than previously thought. That meant the risk per unit dose, or the risk coefficient, was higher, and in fact, it was calculated to be 4×10^{-4} per rem. Based on that increase, the ICRP released a new set of international recommendations in 1990. They recommended limiting radiation exposure to 10 rem over any 5-year period and 5 rem in any one year. The public limit was set at a 100 millirem per year averaged over any 5-year period.

The NCRP released its own new set of national recommendations in 1993[2]. Those limits and the associated risks are listed in Table 2. They relate both to stochastic effects, such as cancer and genetic effects, and to deterministic effects. The present limits for deterministic effects are not much different than the first recommendations: 50 rem per year to any tissue or organ and 15 rem to the lens of the eye to avoid cataract formation. The recommended limits on whole-body doses for stochastic effects, first set at 5 rem per year in

[2] The 1993 NCRP limits on annual radiation doses relate both to stochastic effects, such as cancer and genetic effects, and to deterministic effects, such as cataracts or permanent damage to an organ. Stochastic effects, by definition, arise from random processes. The probability of their occurrence increases with increasing dose, but their severity does not. Moreover, there is no threshold dose below which the risk is zero. In contrast, there is a threshold dose for deterministic effects. That is, doses below the threshold will not kill enough cells to cause dysfunction in a tissue or organ.

1958, are now set at no more than 5 rem in any one year and a lifetime average of no more than 1.5 rem per year.

Table 2. Current Standards and Associated Estimates of Risk (NCRP Report Number 116, 1993)

Category	Annual Limit	Recommended Risk Coefficient	Estimated Risk at the Annual Limit
Occupational annual whole-body limit for stochastic effects	5 rem (stochastic)	4×10^{-4} rem^{-1} (for fatal cancer) 8×10^{-5} rem^{-1} (for severe genetic effects)	2 in 1,000 per year 4 in 10,000 per year
Occupational lifetime limit	1 rem x age (years)	—	3 in 100 at age 70
Occupational annual limit for deterministic effects	15 rem to lens of eye 50 rem to any other organ or tissue system	—	no risk if limits not exceeded
Public annual whole body limit for continuous exposure	100 mrem	5×10^{-4} rem^{-1} (for fatal cancer) 1×10^{-4} rem^{-1} (for severe genetic effects)	1 in 10,000 per year 1 in 100,000 per year
Public annual whole-body limit for infrequent exposure	500 mrem	1×10^{-4} rem^{-1}	1 in 10,000 per year
Negligible individual dose (annual whole-body dose per source or practice)	1 mrem	—	no discernable effects (5 in 10,000,000)

The current limits represent a culmination of intensive epidemiology and radiobiological research. However, there are still many open questions regarding the detailed mechanisms that cause biological effects. What are the relative risks of different types of radiations, acute versus chronic exposures, age of exposure, and chronic exposure to low doses? Those concerns dominate discussions on the future evolution of radiation protection standards.