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Foreword

(This Foreword is not a part of American National Standard Research Reactor Site Evaluation, ANS/ANSI-15.7-1977)

The American Nuclear Society Standards Committee established Subcommittee ANS-15 in the fall of 1970 with the task of preparing a standard for the operation of research reactors. In January, 1972 this charter was expanded to include the multiple tasks of preparing all standards for research reactors. To implement this enlarged responsibility, a number of subcommittee work groups were established to develop standards for consideration and complementary action by Subcommittee ANS-15. ANS-15.7 is one of these work groups.

Work Group ANS-15.7 was formed in 1973 to develop ideas and concepts leading to a draft standard for guidance on siting of a research reactor, taking into account (a) site and facility features, (b) dispersion models, and (c) maximum dose commitments associated with boundaries.

The currently available guidance on reactor siting comes largely if not exclusively from Title 10-CFR-Part 100, Reactor Site Criteria, which was written specifically for nuclear power and test reactors which have significant fission product inventories and energy control systems. At the time of the Atomic Safety and License Appeal Board decision in support of the research reactor license at Columbia University (May 1972) a decision had been made by the Board that Title 10-CFR-Part 100, Reactor Site Criteria, did not apply to research reactors and that there were no other Atomic Energy Commission regulations which specifically defined the criteria for evaluating the effects of a postulated design basis accident at a research reactor. Further, it was stated that Title 10-CFR-Part 20, Standards for Protection Against Radiation, may be overly restrictive in this regard. Hearing board and courts essentially decreed that standards applicable to research reactor site evaluation would be developed.

The original draft of ANS-15.7 utilized definitions and terms similar to Title 10-CFR-Part 100, Reactor Site Criteria. The January, 1975 review by the ANS-15 General Subcommittee revealed that these concepts were confusing and difficult to apply to a much simpler reactor such as a research reactor. This draft standard has attempted to establish definitions, criteria, and depth sufficiently different from those used in familiar Title 10-CFR-Part 100, Reactor Site Criteria, so that definitions and radiation dose parameters will relate to a facility markedly different from a nuclear power plant or test reactor.

The draft standard developmental program reflects the extensive interplay of ANS-15 and the daughter work groups, from the establishment of the working group in 1972 through development of the final draft in January, 1977.

In this process of creating standards against the background of established and varied practices in many operating facilities, it is important to consider that:

a. It is not intended that the standard be used as a demand model for back-fitting purposes.
b. It should be a vital aid for the new owner-agency.
c. It should be helpful for the facility undergoing change or modification.
d. Its thoughtful use by industry should ease the burden of regulatory agencies.

We affirm, further, that the use of any standard of performance, conduct, or excellence is volitional. The decision to use a standard is a management matter, presumably on technical advisement. The institutionalizing of a standard can and almost must be conditional; i.e., high probability exists that some exception or addition will compromise the absolute, unconditional application of a document which was composed to cross lines of functional and material discipline.

This standard is promulgated in the context of these considerations and in the context of a family of related research reactor standards, a work group, and an actively participating subcommittee in an atmosphere of direct exchange of ideas across multi-discipline and multi-system boundaries.

The family of standards and task assignments include:

ANS-15.1	(N378):	Development of Technical Specifications (ANSI N378-1974)
ANS-15.2	(N398):	Quality Verification for Plate-type U-AL Fuel Elements (ANSI N398-1974)
ANS-15.3	(N399):	Records and Reports (ANSI N399-1974)
ANS-15.4	(N380):	Selection and Training of Personnel
ANS-15.6	(N401):	Review of Experiments (ANSI N401-1974)
ANS-15.7	(N379):	Site Evaluation
ANS-15.8	(N402):	Quality Assurance Program Requirements
ANS-15.10	(N440):	Decommissioning
ANS-15.11	(N628):	Radiological Control
ANS-15.12	(N647):	Design Criteria for Systems Controlling Effluents
ANS-15.14	(N700):	Physical Security
ANS-15.15	(N701):	Core Protective Systems
ANS-15.16	(N17.2):	Emergency Planning
ANS-15.17		Fire Protection

Working Group ANS-15.7 of the Standards Committee of the American Nuclear Society had the following membership:

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Frank T. Binford, *Oak Ridge National Laboratory*
Lloyd Bonzon, *Sandia Laboratory Albuquerque*
Tom R. Crites, *Lawrence Livermore Laboratory*
Wade J. Richards, *Argonne National Laboratory*

The membership of ANS-15 at the time of approval of this Standard was:

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The American National Standards Committee N17, Research Reactors, Reactor Physics, and Radiation Shielding, had the following membership at the time it reviewed and approved this Standard:

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R. S. Carter, Secretary

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American Institute of Chemical Engineers	Richard Duffy
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	Alfred M. Perry
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Contents

Research Reactor Site Evaluation

1. Scope

This standard is to be used in evaluating research reactor sites and their associated boundaries.

2. Definitions

For the purpose of this standard, the following words and phrases are defined:

boundaries and zones. The following definitions for boundaries and zones are peculiar to research reactor siting and relate to:

a) the type of authority the reactor chief administrator has over a specific area, and,

b) the potential time required to evacuate a given area to achieve minimum exposure to accident-caused radioactivity.

(1) operations boundary. The operations boundary means the reactor building (or the nearest physical personnel barrier in cases where the reactor building is not a principal physical personnel barrier) where the reactor chief administrator has direct authority over all activities. The area within this boundary shall have prearranged evacuation procedures known to personnel frequenting the area.

(2) rural zone. A rural zone is a sparsely populated but not directly controlled area or neighborhood where evacuation of all personnel can be achieved in less than 2 hours using available resources.

(3) site boundary. The site boundary is that boundary, not necessarily having restrictive barriers, surrounding the operations boundary wherein the reactor administrator may directly initiate emergency activities. The area within the site boundary may be frequented by people unacquainted with the reactor operations.

(4) urban boundary. The urban boundary means the nearest boundary of a densely populated area or neighborhood containing population of such number or in such a location that a complete rapid evacuation is difficult or cannot be accomplished within 2 hours using available resources.

capable fault. A capable fault is a fault which has exhibited one or more of the following characteristics:

(A) Movement at or near the ground surface at least once in the past 35,000 years or more than once in the past 500,000 years. In the absence of data permitting absolute dating, faults with sufficiently recent movement to leave perceptible evidence of surface rupture, surface warping, or offset of geometric features are considered capable faults.

(B) Instrumentally well-determined macroseismicity for a fault located in the continental United States west of the Rocky Mountain front, or in Alaska, Hawaii, or Puerto Rico.

(C) A relationship to a capable fault according to characteristics (A) or (B) such that movement on one could be reasonably expected to be accompanied by movement on the other. (Title 10-CFR — Part 100, "Reactor Site Criteria," Appendix A — "Seismic and Geologic Siting Criteria For Nuclear Power Plants.") [1][1]

design basis accident. A design basis accident (DBA) is a postulated accident used to evaluate the site and the engineered safety features. The DBA describes consequences more serious than those arising from any probable accident for the reactor under evaluation and usually assumes some degree of radionuclide release; that release is used to evaluate population dose commitments.

dose commitment. Dose commitment is that total radiation dose equivalent, internal or external in origin, to the whole body or specified part of the body, that will be received during the 50-year period following the release of radioactive material to the specific environment. Dose quantities that apply to the "whole body" shall also apply to the head and trunk, active blood-forming organs, gonads and lens of the eyes. Dose quantities that apply to "other organs" shall apply to those organs not specified above.

[1]Numbers in brackets refer to corresponding numbers in Section 7, References.