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American Nuclear Society

administrative controls and quality assurance for the operational phase of nuclear power plants

an American National Standard

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Foreword

(This Foreword is not a part of American National Standard Administrative Controls and Quality Assurance for the Operational Phase of Nuclear Power Plants, ANSI/ANS-3.2-1982.)

Preparation for the first edition of this standard commenced in 1969 prior to the establishment of formal quality assurance requirements for the operation of nuclear power plants. Historically, the administrative controls section of Facility Operating License Technical Specifications had contained provisions for meeting many of the requirements that subsequently became identified with quality assurance for operation. It was the original intent of the standard to define administrative controls for this purpose. The standard was completed during a period when the subject of quality assurance was becoming of increasing interest to the nuclear community. The membership of the subcommittee that developed American National Standard Administrative Controls and Quality Assurance for the Operational Phase of Nuclear Power Plants, N18.7-1972 (ANS-3.2) was strongly oriented toward power reactor operation, and developed a document aimed at providing guidance for administrative controls over activities associated with the operation of nuclear power plants. At the same time Subcommittee N45.2, "Nuclear Quality Assurance Standards," of the American National Standards Committee N45, "Reactor Plants and Their Maintenance," was developing quality assurance standards related to design, construction, maintenance, and modification of nuclear power plant structures, systems, and components.

When N18.7-1972 was approved and issued, the U.S. Nuclear Regulatory Commission (NRC) issued its Safety Guide 33, (now Regulatory Guide 1.33) "Quality Assurance Program Requirements (Operation)," endorsing Draft 8 of ANS-3.2 (which later became ANSI N18.7-1972) and American National Standard N45.2-1971, "Quality Assurance Program Requirements for Nuclear Power Plants." This dual endorsement caused some confusion among users and the Executive Committee of the American National Standards Institute (ANSI) Nuclear Technical Advisory Board (now Nuclear Standards Management Board) directed that an ad hoc Task Force, comprising ANS-3 and a representative of ANSI N45.2 Subcommittees, attempt to develop a single standard that could stand alone in defining "Quality Assurance Program Requirements (Operation)." ANSI N18.7-1976 (ANS-3.2) was the result of that effort.

Following the Three Mile Island accident in 1979, ANS-3 undertook a revision of N18.7-1976 to incorporate the administrative "lessons learned" into the standard. During the course of this effort, American National Standard Quality Assurance Program Requirements for Nuclear Power Plants, ANSI/ASME NQA-1-1979, was issued and approved. This standard superseded several of the N45.2 standards which had previously been incorporated by reference into N18.7-1976. A second purpose of this revision of N18.7 was therefore to reflect the issuance of NQA-1.

The Three Mile Island accident identified the need for significant changes in many aspects of the operation of nuclear power plants, and it will be some time before a true industry consensus is reached on the preferred way to implement many of these changes. As a result, it is expected that additional revisions to this standard will be required in the relatively near future. For example, one area where considerable work remains to be done is in the format and content of emergency procedures. At present, the industry is in a transition from procedures which are predominantly "event oriented" (where the operator is expected to diagnose the cause of an emergency early in the sequence and respond accordingly) to those which are predominantly "symptomatic" (where the operator responds to the effects of the emergency with less early emphasis on diagnosing the cause). At this time, it is not possible to write a consensus standard on the best way to handle such procedures. A second such area involves the human factors review of procedures and activities.

The Code of Federal Regulations, Title 10, "Energy," Part 50, "Licensing of Production and Utilization Facilities," Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants," defines the term "quality assurance" as "...all those planned and systematic activities necessary to provide adequate confidence that a structure, system, or component will perform satisfactorily in service." Inherent in this definition is recognition of the fact that quality assurance encompasses activities associated with doing a job correctly as well as verifying and documenting the satisfactory progress and completion of the work. The performance of work is itself the most fundamental aspect of quality assurance in its broadest sense.

On the other hand, the term quality assurance also has been frequently, and quite properly, used to describe the programs, the technical discipline and the organizational unit established to implement special procedural steps to verify and document the satisfactory completion of work. In this context, the term quality assurance (as a technical specialty or as a formal organization) describes a staff support function to assist in the overall goal of the high quality performance of equipment, structures, procedures and personnel.

Historically, quality assurance as an accepted discipline has been associated with manufacturing and construction activities from which it originated as a separate function. It is identified most clearly with systems of checks, audits, inspections, and other forms of verification that can be applied to products that can be examined at various stages of manufacture or construction before they are placed in service; and with the documentation needed to show conformance to requirements and to perform investigations in the event of subsequent malfunction of those products. The nature of manufacturing or construction activities is such that time usually is available or can be taken to perform verification without affecting the quality of the product or activity.

In contrast to potential effects of deficiencies in manufacturing and construction, deficiencies in operating activities can be much more immediate in their effect. For example, it is important that the dynamic aspects of operation be monitored on an essentially continuous basis. Instrumentation for monitoring, control and actuation of safety systems, and observations by, and response from, the operating staff are both extensively used for this purpose in nuclear power plants. In a nuclear power plant employing proper administrative controls and quality assurance practices, the critical appraisal by supervisory personnel of plant operating evolutions, trends in parameters, maintenance, and day-to-day work practices, is the most significant portion of assuring the quality of plant operation (in the broad sense of the term "quality assurance"), whereas quality assurance (as a technical discipline or an organizational unit) of operating activities is associated principally with checking and verifying the adequacy of operating practices and obtaining correction where it is needed. This standard emphasizes that both operating staff and personnel performing other quality assurance functions have important roles in the "...planned and systematic activities..." specified in the Appendix B definition of quality.

This standard discusses requirements for preoperational tests, while recognizing that these tests fall outside the strict definition of the operational phase. This guidance was included because of the frequent heavy involvement of the operations staff in conducting the preoperational tests, and as a response to requirements to implement the operational quality assurance program prior to the start of the operational phase.

In addition to citation of other standards, this standard has made liberal use of wording used in other standards. In some cases applicable sections of other standards have been used verbatim; in others, portions have been paraphrased to indicate more precisely the applicability of the extracted sections to operating activities.

Appended to this foreword is a chart showing the comparison of 10 CFR 50 Appendix B criteria and NQA-1 requirements with the corresponding section of this standard.

This revised standard was prepared by Subcommittee ANS-3, Reactor Operations, of the American Nuclear Society Standards Committee. At the time of the revision, the membership of the Subcommittee was:

H. J. Green, Chairman, ANS-3, *Tennessee Valley Authority*
J. D. Shiffer, Chairman, Ad Hoc Group, *Pacific Gas and Electric Company*
G. C. Andognini, *Arizona Public Service Company*
H. T. Babb, *South Carolina Electric and Gas Company*
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G. K. Whitham, *Argonne National Laboratory*
P. Walzer, *Combustion Engineering, Inc.*

*Members of the ANS-3.2 Ad Hoc group who participated in the preparation of this revision. Mr. R. C. Parker (Tennessee Valley Authority) also served as a member of the ad hoc group. In addition, the chairman would like to acknowledge the assistance of J. J. Benitou (Pacific Gas and Electric Company) toward the completion of this standard.

Comparison Chart
of
10 CFR 50 Appendix B and NQA-1-1979 Requirements
versus
ANSI/ANS-3.2-1982 Requirements

10 CFR 50	NQA-1-1979			ANS-3.2-1982	Comments
Appendix B	Requirement	Supplement	Appendix	Section	
I	1.	1S-1	1A-1	1.	
I	—	—	—	3.1	
I	—	—	—	3.4.2	Refs. ANSI N18.1 (ANS-3.1)
I	—	—	—	3.3	
II	2.	2S-1	2A-1	3.1	
II	—	2S-2	2A-2	3.3	
II	—	2S-3	2A-3	5.1	
II	—	—	—	5.3	
II	—	—	—	3.5	Refs. ANSI N18.1 (ANS-3.1)
II	—	—	—	3.4.2	
III	3.	3S-1	3A-1	5.2.7.2	Refs. ANSI N45.2.11
IV	4.	4S-1	4A-1	5.2.13.1	
V	5.	—	—	5.2.7	Refs. ANSI/ASME N45.2.4, 5, 6, 8, 11
V	—	—	—	5.3	
VI	6.	6S-1	—	5.2.15	
VII	7.	7S-1	7A-1	5.2.13.2	
VIII	8.	8S-1	—	5.2.13.3	
IX	9.	9S-1	—	5.2.18	
IX	—	—	—	5.2.12	
X	10.	10S-1	—	5.2.17	
XI	11.	11S-1	—	5.2.19	
XII	12.	12S-1	—	5.2.16	Refs. ANSI N45.2.4
XIII	13.	13S-1	—	5.2.13.4	Refs. ANSI N45.2.2
XIV	14.	—	—	5.2.6	
XIV	—	—	—	5.2.14	
XV	15.	15S-1	—	5.2.14	
XVI	16.	—	—	5.2.11	
XVII	17.	17S-1	17A-1	5.2.12	Refs. ANSI N45.2.9
XVIII	18.	18S-1	18A-1	4.5	Refs. ANSI N45.2.12

The American Nuclear Society's Nuclear Power Plant Standards Committee (NUPPSCO) had the following membership at the time of its approval of this standard.

J. F. Mallay, Chairman
M. D. Weber, Secretary

Name of Representative	*Organization Represented*
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	(for the Institute of Electrical and Electronics Engineers Inc.)
R. E. Allen (Alt.)	*United Engineers & Constructors, Inc.*
	(for the Institute of Electrical and Electronics Engineers Inc.)
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P. Bradbury	*Westinghouse Advanced Reactor Division*
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