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an American National Standard

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American National Standard Criteria for Accident Monitoring Functions in Light-Water-Cooled Reactors

Secretariat
American Nuclear Society

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Foreword

(This Foreword is not a part of American National Standard Criteria for Accident Monitoring Functions in Light-Water-Cooled Reactors, ANSI/ANS-4.5-1980.)

ANS-4 established Writing Group ANS-4.5 in July 1979 to prepare a standard on Accident Monitoring Instrumentation which would complement other standards. Two primary objectives were to address (1) that instrumentation which permits the operator to monitor expected parameter changes in the accident period, and (2) extended range instrumentation deemed appropriate for the possibility of encountering previously unforeseen events.

This standard provides:

(1) a list of functions to be performed (Section 5, Design Basis)
(2) a framework to determine those variables to be monitored (Section 5, Design Basis)
(3) an identification of two time phases of interest (Section 3, Definitions)
(4) an identification of three variable types (Section 3, Definitions)
(5) a delineation of applicable criteria for the variables to be monitored (Section 6, Criteria)

Draft 3 of this standard was balloted by ANS-4 and reviewed by the American Nuclear Society (ANS) Nuclear Power Plant Standards Committee (NUPPSCO) in October 1979. This process resulted in negative ballots from three ANS-4 members. Drafts 4 and 5 incorporated changes to resolve the ballot and review comments, and draft 5 was balloted by NUPPSCO in February 1980. Review of the writing group ballot comment resolutions was accomplished by NUPPSCO in March, 1980.

A related proposed standard under development, IEEE Std 497, Trial-Use Standard Criteria for Post Accident Monitoring Instrumentation for Nuclear Power Generating Stations, is being revised by the Institute of Electrical and Electronics Engineers, Inc., Nuclear Power Engineering Committee (IEEE/NPEC) Subcommittee 6, to contain specific design criteria for accident monitoring instrumentation. Accordingly, this standard (ANS-4.5) has been revised to delete specific instrumentation design criteria from Section 6, Criteria, and IEEE-497 has been cited for complete treatment of this subject.

The significant issues in the development of this standard have been:

1. The scope of the document in terms of applicability to the control room operator or the plant operator (licensee). The writing group chose a control room operator scope because he has the greater need for information to monitor the course of an accident. Also, the control room operator information needs are better defined, and the plant operator information needs are still evolving. In this standard, the requirements for displayed information are for the purpose of safely shutting down the plant from the control room, but do not include information that may be needed by the plant owner to assess environmental effects. The writing group also decided that the remote shutdown panel is outside the scope of this standard.

2. The pre-planned operator actions based on the accident analyses in Chapter 15 of a plant's Final Safety Analysis Report (FSAR) and other evaluations, and not previously planned operator actions that may be required during unforeseen events. The writing group established Type A instrumentation for the former, and Type B or C instrumentation for the latter.

3. The monitoring of fission product barrier integrity and the potential for breach of a given barrier. The writing group chose monitoring of actual breach for the in-core

fuel clad, reactor coolant system, and primary reactor containment barrier, and the potential for breach of the primary reactor containment barrier since it represents the most direct and immediate threat to the health and safety of the public.

4. The degree of alignment of accident monitoring instrumentation with IEEE Class 1E (ANS Class SC-3) and whether an intermediate class is needed between 1E and non-1E. The writing group chose to define specific criteria for each variable type in lieu of applying a blanket classification such as Class 1E.

5. Whether a list of variables should be included as an appendix to the standard:
   a. a list of only Type C variables
   b. a list of Type A, B, and C variables.

The writing group chose to include Type B and Type C variable recommendations in the standard.

6. The definitions of Types B and C variables and whether these types should be included in the standard. The writing group chose to define Type B and Type C variables in this standard.

7. The definition of Type D variables and whether this type should be included in the standard. Type D variables, monitored to ascertain that the safety systems are operating as designed, are less important than Type A, B and C for accident monitoring, since safety system operation only implies safety function accomplishment. Consequently, Type D variables and instruments are not considered by the writing group to be within the scope of Accident Monitoring Instrumentation, and guidance on the selection of Type D variables and the specifications of appropriate criteria is not given in this standard. This guidance should be provided in standards for design of safety sytems (e.g., IEEE Std 603-1980, Criteria for Safety Sytems for Nuclear Power Generating Stations, American National Standard Nuclear Safety Criteria for the Design of Stationary Pressurized Water Reactor Plants, N18.2-1973 (ANS-51.1), etc). In addition, a new standard is under development, proposed American National Standard for On-Line Monitoring, ANS-4.6, which is also expected to address requirements for Type D variables.

8. In the Nuclear Regulatory Commission (NRC) Regulatory Guide, 1.97, revision 2, Type E variables have been introduced. Type E variables, monitored to determine the magnitude of the release of radioactive materials and for continually assessing such releases, for providing defense in depth, and for diagnosis, have not been included in this standard since the writing group does not consider that these variables are within the scope of accident monitoring instrumentation. However, ANS NUPPSCO is currently determining whether those Type E variables used for emergency planning purposes should be addressed by formation of an ANS-3 writing group in the near future.

9. In subsections 6.2 (Type B variables) and 6.3 (Type C variables), use of "should" and "shall" has been done in the following manner:

(a) "Shall" statements are required to be met, and are structured to present the general requirement in the first sentence in each subsection; for example, in subsection 6.2.2, "The measured variable shall indicate the accomplishment of control of radioactivity in the core."

(b) "Should" statements then follow this first sentence as very stong recommendations to identify the specific variables of interest and their desired display characteristics. Variations in PWR and BWR plant design preclude the use of 'shall" for these sentences; however, the "should" statements convey the intent of the writing group as strong recommendations while still retaining the flexibility to select equivalent or better alternatives for a specific plant in lieu of these recommendations.

10. The writing group selected five "critical safety functions" for Type B variables out of a larger set of plant safety functions that were considered. These critical safety functions are, in the viewpoint of the writing group, the most important safety functions to be accomplished or maintained. Type B variables were then recommended in 6.2 to provide assurance to the control room operator that the five critical safety functions are being accomplished or maintained.

11. In Table 6.1-1, items 5 and 6 designate out-of-service intervals for information display channels. These criteria have been specified to permit periodic equipment maintenance and to designate equipment performance requirements. In item 6, use of the word "none" for Type A and B variables means that deliberate removal-from-service of one of these information display channels is prohibited during Phase I. Once Phase I has begun, an existing out-of-service Type A or Type B variable information display channel should be restored to service as quickly as possible in order to satisfy the single failure criterion requirement in subsection 6.1.7.

12. The variables identified in subsection 6.2.3 may be augmented in a future revisionof this standard, as analyses necessary to determine conditions for inadequate core cooling evolve over time.

The writing group has recognized the need for additional instrumentation during Phase II to provide for primary reactor containment air and water cleanup; however, this instrumentation has not been addressed in this standard. The user of this document must, however, consider this instrumentation in order to properly define the end of Phase II.

Membership of the writing group at the time of its approval of this standard was as follows:

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Additional input has been provided to the writing group by industry, university, and government participants. The writing group is very appreciative of this assistance.

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

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Contents