

**American National Standard
Application Criteria for Programmable Digital Computer
Systems in Safety Systems of Nuclear Power Generating Stations**

Co-Sponsors

**American Nuclear Society and
Institute of Electrical and Electronics Engineers, Inc.**

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Foreword

(This Foreword is not a part of American National Standard Application Criteria for Programmable Digital Computer Systems in Safety Systems of Nuclear Power Generating Stations, ANSI/IEEE-ANS-7-4.3.2-1982.)

This standard establishes application criteria for programmable digital computer systems in safety systems of nuclear power generating stations. These criteria are established to provide a means for promoting safe practices for design and evaluation of safety system performance and reliability. However, adhering to these will not necessarily fully establish the adequacy of any safety system's functional performance and reliability; nonetheless, omission of any of these criteria will, in most instances, be an indication of safety system inadequacy. This standard does not provide specific requirements for preparation or content of software quality assurance plans.

IEEE Std 603-1980, Standard Criteria for Safety Systems for Nuclear Power Generating Stations (Revision of IEEE Std 603-1977) establishes the functional and design criteria for the power, control, and instrumentation portion of safety systems for nuclear power generating stations. P742/ANS-4.3.2, now known as American National Standard Application Criteria for Programmable Digital Computer Systems in Safety Systems of Nuclear Power Generating Stations, ANSI/IEEE-ANS-7-4.3.2-1982, applies to all digital computer systems used in safety systems (for example, multiprocessor distributed systems as well as larger scale single central processor systems). It has been developed to amplify IEEE Std 603-1980 because of the unique nature of digital computer systems — specifically the software. As such, ANSI/IEEE-ANS-7-4.3.2-1982 establishes no additional functional criteria, or design basis requirements. These criteria are covered adequately by American National Standard Design Basis Criteria for Safety Systems in Nuclear Power Generating Stations, ANSI/ANS-4.1-1978, and IEEE Std 603-1980.

In reviewing areas of application criteria that need amplification when digital computers are utilized in safety systems, the following conclusions were reached:

- (1) The criteria established in IEEE Std 603-1980 and supporting standards require no additional amplification for digital computer hardware.
- (2) Because of inherent differences between hardware and software, IEEE Std 603-1980 does require amplification with regard to the method of designing and qualifying the software.
- (3) Because of the high degree of interdependency between the hardware and software, the integration of these components is unique to digital systems, and in this area IEEE Std 603-1980 requires amplification.

Development of this standard began in 1974 under sponsorship of the American Nuclear Society. In 1978, a joint working group was formed that combined members of ANS and the Institute of Electrical and Electronics Engineers, Inc. (IEEE) with a charter to develop a joint standard.

During the development of ANSI/IEEE-ANS-7-4.3.2-1982, the joint working group was greatly concerned with the level of detail and specific requirements that should be included in this standard. It was decided that the standard should not dictate how the system should be implemented since this would force system development in a rigid direction. Recognizing the dynamic nature of digital systems technology, the standard was structured to provide guidance in the application of future digital system technology to safety systems. This standard is not intended to be a detailed design procedure for computer systems engineers in applying digital computers to safety systems nor is it intended to dictate when a digital computer system is re-

quired as part of a safety system. The intent has been generally to amplify IEEE Std 603-1980 and not to restrict unduly the designers or engineers in implementation of digital systems, and also to permit the future application of new digital system technology.

Figure i of the foreword shows the development of the programmable digital computer system in relation to the development of the total safety system.

Figure 1 of the standard illustrates the inter-relationships among the various processes in the application of programmable digital computer systems in safety systems for nuclear power generating stations.

In the development of ANSI/IEEE-ANS-7-4.3.2-1982, the working group specified that verification and validation were essential processes in the development of computer systems utilized in safety systems. Verification and validation are extensions of the concept of testing software to determine that it will perform the correct functions.

Verification is the comparison of the stage-by-stage software development to determine that there is a faithful translation of one stage (such as the design) into the next stage (such as the implementation).

Verification is accomplished in the present state of the art through a communication of concepts, and an understanding of functions, between knowledgeable persons that draw from previous experience and supplementary information. If the translation of one stage to another can be understood by knowledgeable persons, other than the originator, and it is determined that a faithful and accurate translation has been performed, then that stage to stage verification can be considered satisfied. Discrepancies must be documented, and a decision must be made as to what previous stage or stages must be modified (or what action must be taken) to resolve any problems.

Validation assumes that the "Safety System Requirements for Programmable Digital Computer System" is the defining document and provides a comparison with the functions implemented by the computer program in the computer hardware. This validation process provides an overall assurance that the functions specified are implemented in the hardware-software. It also provides assurance that the overall accumulation of the undesired stage to stage side effects have been corrected.

The working group discussed independent verification in detail. Independence is needed:

(1) To meet the quality assurance provisions of Title 10, "Energy," Code of Federal Regulations, Part 50, "Licensing of Production and Utilization Facilities," Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants," Section III, "Design Control," which should be extended to software, as follows:

"The verifying or checking process shall be performed by individuals or groups other than those who performed the original design, but who may be from the same organization."

(2) Because analytic proof of correctness of a complex program is currently impossible, and

(3) To provide an in-depth second analysis of the software requirements and of the tests to confirm that they are met, including abnormal test cases.

One of the greatest advantages of digital computer systems is the flexibility offered by the software system. However, this flexibility has been a liability in balloting this standard. Due to differences in individual perception, one reviewer would classify the standard as "too general," whereas another reviewer would call the same document "too specific." Some reviewers have suggested that the standard should include more specific hardware requirements, including seismic requirements for the digital computer system, bypasses, access to setpoints, manual initiation, capability for test and calibration, etc. These areas have been reviewed by the joint working group during the development of the standard. The consensus of the group was that detailed hardware considerations as well as functional and design criteria for the safety system are fully covered by IEEE Std 603-1980 and other supporting industry standards; e.g., the following American National Standards: Design Basis Criteria for Safety Systems in Nuclear Power Generating Stations, ANSI/ANS-4.1-1978; Nuclear Safety Criteria for the Design of Stationary Pressurized Water Reactor Plants, N18.2-1973 (ANS-51.1); and Nuclear Safety Criteria for the Design of Stationary Boiling Water Reactor Plants, ANSI/ANS-52.1-1978.

The joint working group considers the process of software development to be analogous to that of hardware development. Consequently, the well-established principles and practices of engineering a hardware based system apply similarly to a software-based system.

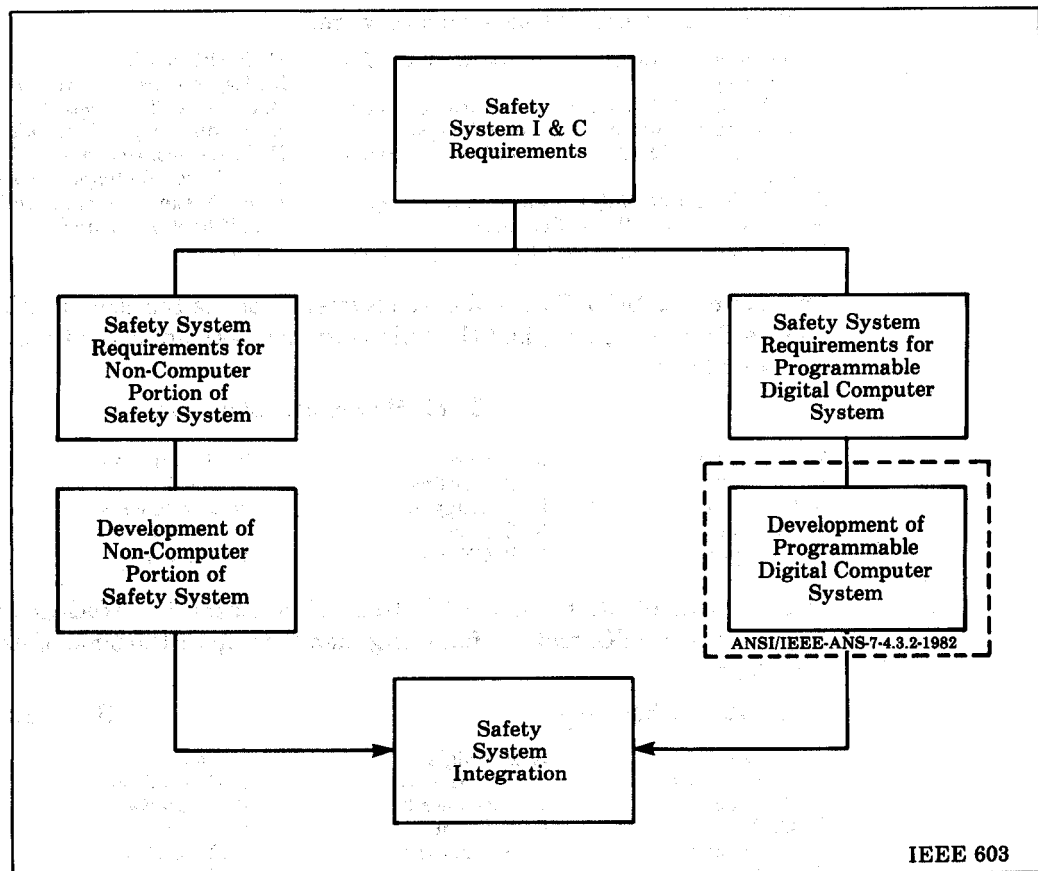


Fig. i
Development of Digital Computer System in Relation to Other I&C Portions of The Safety System