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# **IEEE Standard Criteria for Accident Monitoring Instrumentation for Nuclear Power Generating Stations**

**IEEE Power Engineering Society**

Sponsored by the  
Nuclear Power Engineering Committee



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# IEEE Standard Criteria for Accident Monitoring Instrumentation for Nuclear Power Generating Stations

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**Nuclear Power Engineering Committee  
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**Abstract:** Criteria for variable selection, performance, design, and qualification of accident monitoring instrumentation are established in this standard. Included requirements for display alternatives for accident monitoring instrumentation, documentation of design bases, and use of portable instrumentation.

**Keywords:** accident monitoring, display criteria, selection criteria, type variables.

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# Introduction

(This introduction is not part of IEEE Std 497-2002, IEEE Standard Criteria for Accident Monitoring Instrumentation for Nuclear Power Generating Stations, but is included for information purposes only.)

In the aftermath of an incident at a United States nuclear power plant in March of 1979, a more rigorous approach to developing an accident monitoring system occurred in the United States. This approach culminated in three major sources of requirements for such a system namely:

- ANSI/ANS Std 4.5-1980, Criteria for Accident Monitoring Functions in Light-Water-Cooled Reactors (provided selection and performance criteria)
- IEEE Std 497-1981, IEEE Standard Criteria for Accident Monitoring Instrumentation for Nuclear Power Generating Stations (provided design criteria)
- Regulatory Guide (RG) 1.97, Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant and Environs Conditions During and Following an Accident, Rev 3, May 1983 (prescribed a detailed list of variables to monitor and specified a comprehensive list of design and qualification requirements to be met)

The prescriptive and regulatory-driven nature of RG 1.97 resulted in its becoming the de facto standard for accident monitoring. Although containing many useful approaches and guidance for designing an accident monitoring system, both IEEE Std 497-1981 and ANSI/ANS Std 4.5-1980 became little used and were eventually withdrawn as active standards.

More recently, a need for developing a more flexible consolidated standard arose. The impetus for this need was the desire for increased use of microprocessor based instrumentation systems in both old and the next generation of advanced design nuclear power plants. A need also developed for additional flexibility while modifying past designs and addressing design basis issues. In response to this need, in November of 2000 the Nuclear Power Engineering Committee of IEEE's Power Engineering Society directed its Working Group 6.1 to prepare a new revision of IEEE Std 497-1981.

The new revision was to provide a consolidated source of post-accident monitoring requirements and bases for a new generation of advanced nuclear plant designs. This revision was to also contain guidance and allow a flexible basis for making changes to such systems in older plants. In essence, the revision was to integrate and incorporate applicable requirements from IEEE Std 497-1981, ANSI/ANS Std 4.5-1980, and RG 1.97, as well as to consider state-of-the-art digital design techniques for accident monitoring displays and user feedback from review and use of these three source documents.

Based on this direction, the working group elected to develop an approach to accident monitoring that is both simpler and less prescriptive in nature. That is, instead of providing a list of instrument variables to monitor, the intention would be to provide criteria for how the variables would be selected. Also, instead of up to three possible design and qualification category requirements for each variable type, the intent was to standardize the requirements based on level of importance of the variable type. The result of this effort is IEEE Std 497-2002.

A valuable aspect of this current revision is the criteria provided for advanced instrumentation system designs and design modifications based on modern digital technology. With current technology there are many possibilities for accident monitoring channel configurations that acceptably meet the criteria of this standard. Figure 2 of this standard identifies options from analog signal channels to various combinations of digital signal processing, data validation processing and display afforded by, but not limited to, today's technology. The definitions associated with Figure 1 of this standard are helpful in understanding the options shown in Figure 2. It was also the working group's intention to address the control room displays of information using computer generated displays and calculated values. The criteria presented in this standard provide useful guidance in this area without limiting the types of displays that can be made available to the control room operator.

IEEE Std 497-2002 is arranged to provide six types of accident monitoring criteria as follows:

- How to select and categorize variables. (Clause 4)
- What performance requirements must be met. (Clause 5)
- What design features need to be considered. (Clause 6)
- What aspects of seismic and environmental qualification must be met for each variable type. (Clause 7)
- What display requirements to assure control room operators are properly informed. (Clause 8)
- What quality assurance requirements should apply. (Clause 9).

In addition, an informative annex providing guidance on establishing instrument accuracy requirements has been included.

The following discussion provides guidance and clarification on the use of this standard in the areas of variable selection, establishing design requirements in a simplified way based on level of importance, and qualification clarifications based on function and need.

The selection criteria in this standard allow the defining of five variable types similar to RG 1.97. Type A is accident specific and needed for preplanned operator action, Type B and Type C allow a supervisory overview approach to accident monitoring by allowing a review of critical high level safety functions, Type C additionally allows extended range monitoring of defense-in-depth variables. Type D and Type E allow monitoring of performance of appropriate safety and radiation monitoring systems. The selection criteria, if properly imposed, would result in a list of variables similar to that required by RG 1.97. However, other approaches to implementation of the criteria allow the possibility of other equally acceptable variations. Thus an optimum combination of accident monitoring variables is now more likely and can be defended.

Accident monitoring variable selection must be consistent with the plant specific emergency operating procedures (EOPs) and abnormal operating procedures (AOPs). The variables selected from these procedures need to be the minimum set to assess that safety-related functions are performed and safety systems operate acceptably. It is not intended that the standard apply to instrumentation for contingency actions listed in the EOPs. Also, instrumentation for shutdown from outside the main control room (i.e., remote shutdown) is outside the scope of this standard.

The EOPs derive from Emergency Procedure Guidelines (EPGs), which are developed by the Nuclear Steam Supply System (NSSS) Vendor Owners' Group. For example, such EPGs include functional restoration EPGs or Plant Critical Safety Function Status Trees (depending on which NSSS vendor) for technical emergency procedure input. The working group attempted to use terminology from multiple NSSS vendors for improved universality in this standard.

Accident monitoring instrumentation monitors a large number of variables with a wide variation in the level of importance to the operators. Because of this variation, it would not be reasonable to require that all of the instruments be designed and qualified for all aspects of Class 1E equipment, nor to exclude all instruments from all requirements. Recognizing the potential value to the user, the working group has established the design, display, qualification, and quality requirements for accident monitoring instrumentation in a manner that is consistent with the level of importance of the variables that are being monitored.

The design, display, qualification, and quality criteria of this standard differ from the approach detailed in RG 1.97, which specifies design and qualification requirements for each variable in terms of one of three possible design and qualification categories, that is, Categories 1, 2 and 3. In developing this revision, the working group adopted a philosophy where the qualification requirements for all variables within a type group are either: completely consistent within the group or; consistent with the individual variable's assigned accident monitoring function. This philosophy results in all Type B and Type C variable instruments having the same design, display, qualification, and quality requirements. Type B and Type C

variables do not have any accident specific functions, but instead have the same function for all accidents. For variables categorized as Type A, Type D, or Type E, the design requirements are consistent within each respective group. The qualification (environmental and seismic) requirements applicable to a variable are based on individual variable functional needs and postulated accident conditions at the installed location. Note: Reviews performed by the Working Group confirm that this more simplistic approach (eliminating the need for three design and qualification categories separate from the five variable type groups) results in no new design and qualification requirements above that in place at nuclear plants, which presently comply in an acceptable manner with RG 1.97.

The working group also considered and adopted “lessons learned” from established equipment qualification programs. For example, this standard recognizes that equipment used for accident monitoring may have different qualification requirements based on the time duration relied upon during the accident, the environmental effects present during the postulated accident or the need for that instrument for a particular accident. To illustrate this philosophy, three examples are offered:

*Example 1:* The plant safety analysis identifies that a Type A variable is required to initiate a specific planned manually-controlled action to support a safety-related function for only one specific accident in which no post-accident harsh environment exists. The instrumentation does not have a Type A function for other accidents that produce a harsh environment at the instrument location. The instrumentation for this variable does not require environmental qualification to a harsh environment by the criteria of this standard. However, those Type A variables that are needed to terminate or mitigate an accident producing a harsh environment shall be environmentally qualified for the worst case applicable accident environment.

*Example 2:* High Pressure Coolant Injection (HPCI) system flow rate is a Type D variable used by boiling water reactory (BWR) plant control room operators to indicate and assess the operation of the HPCI system after an accident. The HPCI flow sensor is located in the same room as the HPCI pump, the pump’s steam turbine prime mover, and the associated supply steam piping. The room also contains temperature sensors for steam leak detection. If for illustration purposes, it is assumed that the only source of steam flooding in the room is a pipe break that disables the HPCI pump steam turbine, then it is not necessary to environmentally qualify the HPCI flow sensor instrumentation located in that room for that pipe break steam environment. This conclusion is arrived at by observing that the HPCI system is not required to operate during this accident scenario (because it is disabled) and, as a result, there is no performance to assess. Additionally, the room’s leak detection system provides information on the system’s availability and status.

*Example 3:* A Type E radiation monitor is mounted on a non-seismically designed plant effluent vent. This vent path can be isolated further upstream by a seismically designed and constructed damper and the position of this damper is indicated in the control room. The requirements of this standard may not require that this Type E variable instrumentation be seismically qualified.

In conclusion, although written primarily for new plant designs, existing plants may also find useful guidance and applicable criteria in this standard. The use of applicable plant procedures to determine the requirements of the accident monitoring instrumentation provides the necessary flexibility for useful design criteria. This standard can be used to address the necessary changes to the plant configuration that inevitably occur over the operating life of any plant. This standard also offers some advantages in clarifying when environmental and seismic qualification would be required and establishing design criteria for digital equipment upgrades involving accident monitoring instrument channels.

## Future Work

As the use of computers in the nuclear plant is a dynamic area of design, the working group intends to keep this area as one of its ongoing future tasks.

The treatment of accident monitoring instrumentation for severe accidents should be incorporated into this standard in the future. It is the working group's philosophy that sufficient instrumentation should exist to inform operators of the status of the three fission product barriers at all times. Instrumentation needed to monitor plant conditions during a severe accident must perform its function for the time period needed in the best-estimate (using realistic assumptions) environmental conditions of the severe accident (e.g., pressure, temperature, humidity, radiation) for which the instrument is relied upon to function.

Another area that the group believes should be incorporated into the standard in the future is the instrumentation that is used for determining plant emergency classification levels (such as notification of unusual event, alert, site area emergency, and general emergency).

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This standard was prepared by Subcommittee 6 Working Group 6.1 (Post Accident Monitoring) of the Nuclear Power Engineering Committee of IEEE's Power Engineering Society. At the time this standard was balloted, Working Group 6.1 had the following membership:

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# Contents

|      |                                           |    |
|------|-------------------------------------------|----|
| 1.   | Overview.....                             | 1  |
| 1.1  | Scope.....                                | 1  |
| 1.2  | Purpose.....                              | 1  |
| 1.3  | Application.....                          | 1  |
| 2.   | References.....                           | 2  |
| 3.   | Definitions.....                          | 2  |
| 4.   | Selection criteria .....                  | 5  |
| 4.1  | Type A variables .....                    | 5  |
| 4.2  | Type B variables .....                    | 5  |
| 4.3  | Type C variables .....                    | 6  |
| 4.4  | Type D variables .....                    | 6  |
| 4.5  | Type E variables .....                    | 6  |
| 4.6  | Documentation of selection bases.....     | 6  |
| 5.   | Performance criteria.....                 | 8  |
| 5.1  | Range .....                               | 8  |
| 5.2  | Accuracy .....                            | 8  |
| 5.3  | Response time .....                       | 8  |
| 5.4  | Required instrumentation duration .....   | 9  |
| 5.5  | Reliability.....                          | 9  |
| 5.6  | Performance assessment documentation..... | 9  |
| 6.   | Design criteria.....                      | 9  |
| 6.1  | Single failure.....                       | 9  |
| 6.2  | Common cause failure .....                | 10 |
| 6.3  | Independence and separation .....         | 11 |
| 6.4  | Isolation.....                            | 11 |
| 6.5  | Information ambiguity .....               | 11 |
| 6.6  | Power supply.....                         | 12 |
| 6.7  | Calibration.....                          | 12 |
| 6.8  | Testability .....                         | 12 |
| 6.9  | Direct measurement .....                  | 13 |
| 6.10 | Control of access.....                    | 13 |
| 6.11 | Maintenance and repair.....               | 13 |
| 6.12 | Minimizing measurements.....              | 13 |
| 6.13 | Auxiliary supporting features .....       | 13 |
| 6.14 | Portable instruments.....                 | 13 |
| 6.15 | Documentation of design criteria.....     | 13 |
| 7.   | Qualification criteria .....              | 14 |
| 7.1  | Type A variables .....                    | 14 |
| 7.2  | Type B variables .....                    | 14 |
| 7.3  | Type C variables .....                    | 14 |