U.S. NUCLEAR REGULATORY COMMISSION

Revision 4
December 2005

REGULATORY GUIDE

OFFICE OF NUCLEAR REGULATORY RESEARCH

Prepublication

REGULATORY GUIDE 1.97

(*Draft was issued as DG-1128, dated June 2005*)

CRITERIA FOR ACCIDENT MONITORING INSTRUMENTATION FOR NUCLEAR POWER PLANTS

A. INTRODUCTION

The U.S. Nuclear Regulatory Commission (NRC) developed this regulatory guide to describe a method that the NRC staff considers acceptable for use in complying with the agency's regulations with respect to satisfying criteria for accident monitoring instrumentation in nuclear power plants. Specifically, the method described in this regulatory guide relates to General Design Criteria 13, 19, and 64, as set forth in Appendix A to Title 10, Part 50, of the *Code of Federal Regulations* (10 CFR Part 50), "Domestic Licensing of Production and Utilization Facilities":

- Criterion 13, "Instrumentation and Control," requires operating reactor licensees to provide instrumentation to monitor variables and systems over their anticipated ranges for accident conditions as appropriate to ensure adequate safety.
- Criterion 19, "Control Room," requires operating reactor licensees to provide a control room from which actions can be taken to maintain the nuclear power unit in a safe condition under accident conditions, including loss-of-coolant accidents (LOCAs). In addition, operating reactor licensees must provide equipment (including the necessary instrumentation), at appropriate locations outside the control room, with a design capability for prompt hot shutdown of the reactor.
- Criterion 64, "Monitoring Radioactivity Releases," requires operating reactor licensees to provide the means for monitoring the reactor containment atmosphere, spaces containing components to recirculate LOCA fluids, effluent discharge paths, and the plant environs for radioactivity that may be released as a result of postulated accidents.

In addition, Subsection (2)(xix) of 10 CFR 50.34(f), "Additional TMI-Related Requirements," requires operating reactor licensees to provide adequate instrumentation for use in monitoring plant conditions following an accident that includes core damage.

This revision of Regulatory Guide 1.97 represents an ongoing evolution in the nuclear industry's thinking and approaches with regard to accident monitoring systems for the Nation's nuclear power plants. Specifically, this revision endorses (with certain clarifying regulatory positions specified in Section C of this guide) the "IEEE Standard Criteria for Accident Monitoring Instrumentation for Nuclear Power Generating Stations," which the Institute of Electrical and Electronics Engineers (IEEE) promulgated as IEEE Std. 497-2002.[1]

This revised regulatory guide is intended for licensees of new nuclear power plants.[2] Previous revisions of this regulatory guide remain in effect for licensees of current operating reactors,[2] who are unaffected by this revision. (See the discussion of regulatory position #1 in Section C of this guide regarding the applicability of IEEE Std. 497-2002 for current operating reactors.)

In general, information provided by regulatory guides is reflected in the NRC's "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants" (NUREG-0800).[3] The NRC's Office of Nuclear Reactor Regulation (NRR) uses the Standard Review Plan (SRP) to review applications to construct and operate nuclear power plants. Chapter 7, "Instrumentation and Controls," and its Branch Technical Position HICB-10, "Guidance on Application of Regulatory Guide 1.97," of the SRP will require updates for consistency with this revision of Regulatory Guide 1.97.

Any information collections mentioned in this regulatory guide are established as requirements in 10 CFR Part 50, which provides the regulatory basis for this guide. The Office of Management and Budget (OMB) has approved those information collection requirements under OMB control number 3150-0011. The NRC may neither conduct nor sponsor, and a person is not required to respond to, a request for information or an information collection requirement unless the requesting document displays a currently valid OMB control number.

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1 IEEE publications may be purchased from the IEEE Service Center, which is located at 445 Hoes Lane, Piscataway, NJ 08855 [http://www.ieee.org, phone (800) 678-4333].

2 The terms "new nuclear power plant" and "new plant" refer to any nuclear power plant for which the licensee obtained an operating license after the NRC issued Revision 4 of Regulatory Guide 1.97. The terms "current operating reactor" and "current plant" refer to any nuclear power plant for which the licensee obtained an operating license before the NRC issued Revision 4 of Regulatory Guide 1.97.

3 Copies are available at current rates from the U.S. Government Printing Office, P.O. Box 37082, Washington, DC 20402-9328 [telephone (202) 512-1800], or from the National Technical Information Service (NTIS), 5285 Port Royal Road, Springfield, Virginia 22161 [http://www.ntis.gov, telephone (703) 487-4650]. Copies are available for inspection or copying for a fee from the NRC's Public Document Room (PDR), which is located at 11555 Rockville Pike, Rockville, Maryland; the PDR's mailing address is USNRC PDR, Washington, DC 20555-0001. The PDR can also be reached by telephone at (301) 415-4737 or (800) 397-4205, by fax at (301) 415-3548, and by email to PDR@nrc.gov.

B. DISCUSSION

In the aftermath of the accident at Three Mile Island, Unit 2 (TMI-2), in 1979, the United States adopted a more rigorous approach for accident monitoring systems, which resulted in three major sources of related requirements:

(1) ANSI/ANS-4.5-1980, "Criteria for Accident Monitoring Functions in Light-Water-Cooled Reactors,"[4] delineated criteria for determining the variables that the control room operator should monitor to ensure safety during an accident and the subsequent long-term stable shutdown phase. The American National Standards Institute (ANSI) promulgated this standard, which was developed by the American Nuclear Society (ANS) Standards Committee, Subcommittee ANS-4, Writing Group 4.5. In so doing, ANSI and ANS sought to address (1) instrumentation that permits operators to monitor expected parameter changes during an accident, and (2) extended-range instrumentation deemed appropriate for previously unforeseen events. As the source for specific instrumentation design criteria, ANSI/ANS-4.5-1980 referenced the draft IEEE Std. 497-1977, "IEEE Trial-Use Standard Criteria for Accident Monitoring Instrumentation for Nuclear Power Generating Stations,"[5] which IEEE subsequently issued as IEEE Std. 497-1981.[5]

(2) IEEE Std. 497-1981, "IEEE Standard Criteria for Accident Monitoring Instrumentation for Nuclear Power Generating Stations," provided the relevant instrumentation design criteria.

(3) Revision 3 of Regulatory Guide 1.97, "Instrumentation for Light-Water-Cooled Nuclear Power Plants To Assess Plant and Environs Conditions During and Following an Accident,"[6] dated May 1983, prescribed a detailed list of variables to monitor, and specified a comprehensive list of design and qualification criteria to be met.

Given its prescriptive nature, Revision 3 of Regulatory Guide 1.97 quickly became the de facto standard for accident monitoring, and both ANSI/ANS-4.5-1980 and IEEE Std. 497-1981 fell out of use and were subsequently withdrawn as active standards. Nonetheless, Revision 3 of Regulatory Guide 1.97 has become outdated, in that it does not provide criteria for advanced instrumentation system designs based on modern digital technology. Revision 3 also does not address the need for technology-neutral guidance for licensing new plants. In addition, the guidance should be less prescriptive and based on the accident management functions of the individual variable types.

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4 Copies may be obtained from the American Nuclear Society, which is located at 555 North Kensington Avenue, La Grange Park, Illinois 60525 [http://www.ans.org, phone (708) 352-6611].

5 IEEE publications may be purchased from the IEEE Service Center, which is located at 445 Hoes Lane, Piscataway, NJ 08855 [http://www.ieee.org, phone (800) 678-4333].

6 Copies are available at current rates from the U.S. Government Printing Office, P.O. Box 37082, Washington, DC 20402-9328 [telephone (202) 512-1800], or from the National Technical Information Service (NTIS), 5285 Port Royal Road, Springfield, Virginia 22161 [http://www.ntis.gov, telephone (703) 487-4650]. Copies are available for inspection or copying for a fee from the NRC's Public Document Room (PDR), which is located at 11555 Rockville Pike, Rockville, Maryland; the PDR's mailing address is USNRC PDR, Washington, DC 20555-0001. The PDR can also be reached by telephone at (301) 415-4737 or (800) 397-4205, by fax at (301) 415-3548, and by email to PDR@nrc.gov.

With the increased use of digital instrumentation systems in advanced nuclear power plant designs, the nuclear industry came to recognize a need to develop a consolidated standard that was more flexible than Revision 3 of Regulatory Guide 1.97. Instead of prescribing the instrument variables to be monitored (as was the case in Revision 3), the industry recognized the advantage of providing performance-based criteria for use in selecting variables. Similarly, rather than providing design and qualification criteria for each variable category, the industry sought to standardize the criteria based on the accident management functions of the given type of variable. These efforts resulted in the development of IEEE Std. 497-2002, "IEEE Standard Criteria for Accident Monitoring Instrumentation for Nuclear Power Generating Stations,"[7] by the IEEE Power Engineering Society, Nuclear Power Engineering Committee, Subcommittee 6, Working Group 6.1, "Post-Accident Monitoring."

Unlike its predecessor, IEEE Std. 497-2002 establishes flexible, performance-based criteria for the selection, performance, design, qualification, display and quality assurance of accident monitoring variables. As such, these variables are the operators' primary sources of accident monitoring information. In some instances, additional variables which provide backup or diagnostic information may exist; however, these backup and diagnostic variables, which are not considered primary sources of information, need not be classified in accordance with the variable types in IEEE Std. 497-2002, and they need not meet the criteria in this guide.

Clause 8.1.2 of IEEE Std. 497-2002 cites several industry codes and standards for human factors criteria. The NRC provides additional guidance in NUREG-0700, "Human-System Interface Design Review Guideline: Review Methodology and Procedures"[8]; NUREG-0711, "Human Factors Engineering Program Review Model"[8]; and Chapter 18, "Human Factors Engineering,"of the NRC's Standard Review Plan (NUREG-0800).[8]

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8 Copies are available at current rates from the U.S. Government Printing Office, P.O. Box 37082, Washington, DC 20402-9328 [telephone (202) 512-1800], or from the National Technical Information Service (NTIS), 5285 Port Royal Road, Springfield, Virginia 22161 [http://www.ntis.gov, telephone (703) 487-4650]. Copies are available for inspection or copying for a fee from the NRC's Public Document Room (PDR), which is located at 11555 Rockville Pike, Rockville, Maryland; the PDR's mailing address is USNRC PDR, Washington, DC 20555-0001. The PDR can also be reached by telephone at (301) 415-4737 or (800) 397-4205, by fax at (301) 415-3548, and by email to PDR@nrc.gov.

Clause 6.2 of IEEE Std. 497-2002 states, in part, that the use of identical software in redundant instrumentation channels is acceptable, provided that the licensee conducts an analysis to demonstrate defense-in-depth against common-mode software failure. The NRC provides related guidance in Branch Technical Position HICB-19, "Guidance for Evaluation of Defense-in-Depth and Diversity in Digital Computer-Based Instrumentation and Control Systems,"[9] as detailed in Chapter 7 of the NRC's Standard Review Plan (NUREG-0800).

In addition, IEEE Std. 497-2002 includes two informative annexes:

- Annex A provides general guidance regarding "Accident Monitoring Instrument Channel Accuracy." In that annex, Clause A.2 provides guidance on accuracy requirement groupings according to how control room personnel should use the displayed functions, while Clause A.3 provides typical accuracy requirements. Specifically, Clause A.3 states, in part, "Historically, the required accuracy for instrument channels relied upon to monitor containment pressure and hydrogen concentration has been ±10 percent of full span." However, the NRC staff notes that this example may not be applicable to all nuclear power plants. Traditionally, the required accuracy of accident monitoring instrument channels is established based on the assigned function and the plant's safety analysis and licensing basis.
- Annex B, "Bibliography," lists the references cited in the standard, and provides sufficient detail for users to obtain further information regarding specific aspects of the standard.

9 Copies are available at current rates from the U.S. Government Printing Office, P.O. Box 37082, Washington, DC 20402-9328 [telephone (202) 512-1800], or from the National Technical Information Service (NTIS), 5285 Port Royal Road, Springfield, Virginia 22161 [http://www.ntis.gov, telephone (703) 487-4650]. Copies are available for inspection or copying for a fee from the NRC's Public Document Room (PDR), which is located at 11555 Rockville Pike, Rockville, Maryland; the PDR's mailing address is USNRC PDR, Washington, DC 20555-0001. The PDR can also be reached by telephone at (301) 415-4737 or (800) 397-4205, by fax at (301) 415-3548, and by email to PDR@nrc.gov.

C. REGULATORY POSITION

This regulatory guide endorses IEEE Std. 497-2002, "IEEE Standard Criteria for Accident Monitoring Instrumentation for Nuclear Power Generating Stations," as an acceptable method for providing instrumentation to monitor variables for accident conditions, subject to the following regulatory positions:

(1) *If a current operating reactor licensee voluntarily converts to the criteria in Revision 4 of this guide, the licensee should perform the conversion on the plant's entire accident monitoring program to ensure a complete analysis.*

Regulatory position #1 clarifies the applicability of IEEE Std. 497-2002 for current operating reactors. Clause 1.1 of IEEE Std. 497-2002 states that the standard is intended for new plants, although current plants may find its guidance useful in performing design-basis evaluations or implementing design modifications. Having carefully considered the applicability and usefulness of the new standard, the NRC staff recognizes that current operating reactors could be interested in converting to IEEE Std. 497-2002. In this context, conversion means adapting *the plant's entire accident monitoring program* from a given plant's current licensing basis (namely Revision 2 or 3 of this guide), to the guidance in Revision 4 of this guide. This adaptation could include physical changes (e.g., replacing an instrument), licensing changes (e.g., technical specification changes), or both for each variable. The staff also recognizes that Revisions 3 and 4 of this guide differ in several ways, including variable type definitions and associated criteria, removal of design and qualification categories, removal of prescriptive tables of monitored variables, analysis required to produce the necessary design-basis documentation, and related changes in licensing basis and/or commitments. These differences could involve modifications to existing instrumentation and could have significant cost implications for current operating reactor licensees who decide to convert to the new standard under Revision 4 of this guide. Revision 4 is primarily intended for licensees of new nuclear power plants. However, the NRC staff sees no technical reason to prohibit a current operating reactor licensee from voluntarily making this conversion. Partial conversions (i.e., conversions only performed on particular variables or systems) are not recommended because of the potential for loss of variables or interactions with other variables without a complete analysis in accordance with this guide.

(2) *Modify the first sentence in the second paragraph of Clause 6.7, as follows:*

"Means shall be provided for validating instrument calibration during the accident."

Regulatory position #2 modifies the requirement of IEEE Std. 497-2002, as it relates to instrumentation calibration during an accident. Clause 6.7 of IEEE Std. 497-2002 requires licensees to provide the means to calibrate instrumentation during an accident, and Clause 6.11 requires licensees to consider the selection and location of instrumentation with respect to potential inaccessibility during an accident. Plants should strategically locate instruments to ensure that they are readily accessible for maintenance. However, the NRC staff recognizes that some instruments (e.g., in-line sensors and area monitors) must be located in areas that are not accessible during an accident. Furthermore, recalibration is one of the four methods stated in Clause 6.7, but the only method of "maintaining" instrument calibration. In many situations, it is not possible to recalibrate instrumentation during an accident due to environmental conditions at the instrument location. The other three methods stated in Clause 6.7 cannot be used to maintain instrument calibration, but rather can be used to verify that the instrument has not excessively deviated from calibration. Consequently, licensees should provide means for validating instrument calibration during the accident.

(3) *The range criteria for Type C variables (paragraph 2 of Clause 5.1) should include the basis for the expanded ranges as follows:*

"The range for Type C variables shall encompass those limits that would indicate a breach in a fission product barrier. These variables shall have expanded ranges and a source term that consider a damaged core (see NUREG-0660). For example, ..."

Regulatory position #3 clarifies the requirement to provide expanded ranges for Type C variables, which Clause 4.3 of IEEE Std. 497-2002 describes as those "that provide the most direct indication of the integrity of the three fission product barriers and provide the capability for monitoring beyond the normal operating range." Clause 5.1 of the standard adds, "the range for Type C variables shall encompass, with margin, those limits that would indicate a breach in a fission product barrier." In a related provision in 10 CFR 50.34(f)(2)(xix), the NRC requires licensees to provide instrumentation to monitor plant conditions following an accident that includes core damage. The underlying basis for this regulation, documented in NUREG-0660, "NRC Action Plan Developed as a Result of the TMI-2 Accident,"[10] was that licensees should provide instrumentation "with expanded ranges and a source term that considers a damaged core capable of surviving the accident environment in which it is located for the length of time its function is required." To include the basis for the expanded range (from NUREG-0660), licensees should modify the range criteria for Type C variables (paragraph 2 of Clause 5.1), as stated in regulatory position #3.

(4) *Modify the last sentence in Clause 4.1 as follows:*

"Type A variables include those variables that are associated with contingency actions that are within the plant licensing basis and may be identified in written procedures."

Modify the last sentence in Clause 1.3, as follows:

"This standard also does not apply to instrumentation required to support plant shutdown from outside the control room."

Regulatory position #4 modifies the application of the term "contingency actions," which Clause 3.6 of IEEE Std. 497-2002 defines as "alternative actions taken to address unexpected responses of the plant or conditions beyond its licensing basis (for example, actions taken for multiple equipment failures)." Clause 1.3 uses this term in defining the application of IEEE Std. 497-2002, while Clause 4.1 uses it in defining selection criteria for Type A variables. The staff agrees with the criteria in these clauses, except where they exclude contingency actions. Contingency actions were excluded from the scope of Revision 3 of this guide, but neither Revision 3 nor its endorsed standard provided a definition of the term "contingency action." NSSS vendors have not used this term consistently in EPGs for current plant designs and, therefore, the staff recommends considering contingency actions in accordance with the modified criteria in Clause 4.1. Furthermore, Revision 3 provided a prescriptive list of variables to monitor, whereas this revision provides a non-prescriptive, performance-based approach to variable selection. Thus, in this performance-based guide, the staff cannot endorse the carte blanche exclusion of contingency actions from the selection criteria (especially those associated with plant-specific operating procedures or guidelines). Rather, the scope of instruments that could potentially be selected for accident monitoring (based on the selection criteria) should initially be as encompassing

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as possible. Then, in the process of selecting the actual list of variables to be monitored, licensees could screen out instruments associated with contingency actions that take place beyond the plant's licensing basis.

(5) *The number of measurement points should be sufficient to adequately indicate the variable value.*

Regulatory position #5 provides guidance concerning the number of measurement points for each variable, which IEEE Std. 497-2002 does not mention (with the exception of redundancy requirements). In general, the number of measurement points should be sufficient to adequately indicate the variable value (e.g., containment temperature may require spatial distribution of several measurement points).

(6) *If the NRC's regulations incorporate an industry code or standard referenced in Clause 2 of IEEE Std. 497-2002, licensees and applicants must comply with that code or standard as set forth in the regulations. Similarly, if the NRC staff has endorsed a referenced code or standard in a regulatory guide, that code or standard constitutes an acceptable method for use in meeting the related regulatory requirement as described in the regulatory guide(s). By contrast, if a referenced code or standard has neither been incorporated into the NRC's regulations nor been endorsed in a regulatory guide, licensees and applicants may consider and use the information in the referenced code or standard, if appropriately justified, consistent with current regulatory practice.*

(7) *Modify paragraph (c) of Clause 5.4, as follows:*

"The operating time for Type C variable instrument channels shall be at least 100 days or the duration for which the measured variable is required by the plant's LBD."

Regulatory position #7 modifies the required instrument duration for Type C variables from "at least 100 days" to include cases where the plant's LBD defines a different operating time. The plant's LBD provides an appropriate basis for determining the operating time for Type C variables and is consistent with the required instrument durations for other variable types. Consequently, licensees may specify the Type C variable operating time based on the plant's LBD.

(8) *Modify Clause 5.4 to replace the term "post-event operating time" with "operating time."*

The term "post-event operating time" implies that the plant is in a controlled condition (the event has been mitigated) when the instrumentation is first required to function. This is inconsistent with the criteria for selection of accident monitoring variables, as the variables are derived from actions based on plant procedures (e.g., AOPs, EOPs, and EPGs). The actions described in these procedures encompass conditions during accident mitigation, as well as when the plant is in a controlled condition. The operating time for each variable is determined by the plant's LBD and should not imply that they are only required during the "post-event" phase of accident management. Consequently, licensees should consider the plant LBD's operating time for each variable when determining the required instrument duration.

D. IMPLEMENTATION

The purpose of this section is to provide information to applicants and licensees regarding the NRC staff's plans for using this regulatory guide. No backfitting is intended or approved in connection with the issuance of this guide.

Except in cases in which an applicant or licensee proposes or has previously established an acceptable alternative method for complying with specified portions of the NRC's regulations, the methods described in this guide will be used in evaluating (1) submittals in connection with applications for construction permits, design certifications, operating licenses, and combined licenses, and (2) submittals from operating reactor licensees who voluntarily propose to initiate system modifications if there is a clear nexus between the proposed modifications and the subject for which guidance is provided herein.

REGULATORY ANALYSIS

1. Problem

In the aftermath of the accident at TMI-2, in 1979, the United States adopted a more rigorous approach for accident monitoring systems, which resulted in three major sources of related requirements:

(1) ANSI/ANS-4.5-1980, "Criteria for Accident Monitoring Functions in Light-Water-Cooled Reactors"

(2) IEEE Std. 497-1981, "IEEE Standard Criteria for Accident Monitoring Instrumentation for Nuclear Power Generating Stations"

(3) Revision 3 of Regulatory Guide 1.97, "Instrumentation for Light-Water-Cooled Nuclear Power Plants To Assess Plant and Environs Conditions During and Following an Accident"

Because Revision 3 of Regulatory Guide 1.97 prescribed a detailed list of variables to monitor, and specified a comprehensive list of design and qualification criteria to be met, it quickly became the de facto standard for accident monitoring, and both ANSI/ANS-4.5-1980 and IEEE Std. 497-1981 fell out of use and were subsequently withdrawn as active standards. Nonetheless, Revision 3 of Regulatory Guide 1.97 has become outdated, in that it does not provide criteria for advanced instrumentation system designs based on modern digital technology. Revision 3 also does not address the need for technology-neutral guidance for licensing new plants. In addition, the guidance should be less prescriptive and based on the accident management functions of the individual variable types.

With the increased use of digital instrumentation systems in advanced nuclear power plant designs, the nuclear industry came to recognize a need to develop a consolidated standard that was more flexible than Revision 3 of Regulatory Guide 1.97. Instead of prescribing the instrument variables to be monitored (as was the case in Revision 3), the industry recognized the advantage of providing performance-based criteria for use in selecting variables. Similarly, rather than providing design and qualification criteria for each variable category, the industry sought to standardize the criteria based on the accident management functions of the given type of variable. These efforts resulted in the development of IEEE Std. 497-2002, "IEEE Standard Criteria for Accident Monitoring Instrumentation for Nuclear Power Generating Stations."

2. Technical Approach

IEEE Std. 497-2002 is an updated national consensus standard that reflects the current state of technology. A valuable aspect of this revised standard is that the criteria provided for advanced instrumentation system designs and design modifications are based on modern digital technology. This allows numerous possibilities for accident monitoring channel configurations that can meet the criteria of IEEE Std. 497-2002. The standard also provides useful guidance regarding computer-generated control room displays and calculated values, without limiting the types of displays that can be made available.

The IEEE Std. 497-2002 criteria for selection, performance, design, qualification, display, and quality assurance are similar to those defined in Revision 3 of Regulatory Guide 1.97. In addition, IEEE Std. 497-2002 integrates requirements from the aforementioned standards and guides, and provides additional criteria and guidance for use of digital instrumentation systems.

In developing Revision 4 of Regulatory Guide 1.97, the NRC staff considered the following alternative approaches:

(1) **Take no action**. This approach would require licensees of new plants to justify a large number of deviations from the guidance provided in Revision 3 of this guide, because advanced plant designs require different accident prevention and mitigation systems and functions than the current generation of light-water reactors. Moreover, without revised guidance, additional deviations may be required in numerous applications where digital instrumentation systems are used. This approach provides no added value, and would require additional staff resources to evaluate deviations on a case-by-case basis. As a result, the staff did not select this approach.

(2) **Revise Regulatory Guide 1.97 to endorse IEEE Std. 497-2002 and incorporate deviations, clarifications, and rule changes that the NRC has previously approved for current operating reactor licensees**. This approach would require the NRC staff to endorse IEEE Std. 497-2002 for both new plants and current operating reactors, because the previously endorsed standard (ANSI/ANS-4.5-1980) has been withdrawn. This approach would also require the NRC to carry forward and update Tables 1–3 and the associated discussion and regulatory positions from Revision 3 of this guide for current operating reactors, while endorsing the IEEE Std. 497-2002 criteria for new plants. In so doing, this approach would provide updated guidance (for both new plants and current operating reactors) in Regulatory Guide 1.97, which the nuclear industry recognizes as the sole source of accident monitoring criteria. However, the resulting regulatory guide would require different sets of criteria for new plants and current operating reactors, which could lead to ambiguities as to the plants (or class of plants) to which the criteria apply. Furthermore, the information carried forward and updated from Revision 3 would not provide any new guidance for current operating reactors and, hence, revising the guide would not yield any benefit. As a result, the staff did not select this approach.

(3) **Develop a new regulatory guide to endorse IEEE Std. 497-2002 for new plants, but leave Regulatory Guide 1.97 at Revision 3 for current operating reactors**. This approach would require distinct guidance for current operating reactor licensees who use Revisions 2 or 3 of Regulatory Guide 1.97, which would differ from the new regulatory guide that endorses IEEE Std. 497-2002 for new plants. As such, this approach has two disadvantages. First, the nuclear industry considers Regulatory Guide 1.97 to be synonymous with accident monitoring criteria. Developing a new regulatory guide for accident monitoring criteria for new plants would require the industry to refer to a second guide with a distinct numeric designation. While this disadvantage may seem minor, it creates an undesirable and potentially confusing situation for both licensees and regulators. Second, and more importantly, many regulatory documents (CFR, regulatory guides, NUREG-series reports, etc.) refer to Regulatory Guide 1.97 for accident monitoring criteria. Developing a second regulatory guide would require the NRC to revise each of these regulatory documents to reference both Regulatory Guide 1.97 and the new regulatory guide when referring to accident monitoring criteria. This would also create a temporary disconnect between the regulatory documents and the new regulatory guide for new plants until the NRC can revise all of these regulatory documents. This approach is achievable and would clearly distinguish the guidance for new plants and current operating reactors; however, it would be extremely labor- and time-intensive. As a result, the disadvantages and cost of implementing this approach far outweigh its limited added value.

(4) **Revise Regulatory Guide 1.97 to endorse IEEE Std. 497-2002 for new plants and remove all previous guidance for current operating reactors**. Since current operating reactors have licensing commitments to Revision 2 or 3 of Regulatory Guide 1.97, they are not required to follow any subsequent revisions. As a result, this approach would have no impact on current operating reactor licensees because Revision 4 of Regulatory Guide 1.97 would be intended for new plants. Current operating reactor licensees would simply have to rely on Branch Technical Position HICB-10 of the Standard Review Plan (NUREG-0800) to consolidate any generic approved deviations, as is currently being done. Nonetheless, this approach would concisely provide the most current acceptable method for use in meeting the NRC's accident monitoring regulations under the industry-recognized name of Regulatory Guide 1.97. As a result, the staff concluded that this is the preferable solution to the problem.

Revision 4 of Regulatory Guide 1.97 is primarily intended for new plants. The NRC staff recognizes that current operating reactors could be interested in converting accident monitoring variables from their current licensing basis (namely Revision 2 or 3 of this guide) to the guidance in Revision 4 of this guide. The staff also recognizes that Revisions 3 and 4 of this guide differ in several ways, including variable type definitions and associated criteria, removal of design and qualification categories, removal of prescriptive tables of monitored variables, analysis required to produce the necessary design-basis documentation, and related changes in licensing basis and/or commitments. These differences could involve modifications to existing instrumentation and could have significant cost implications for current operating reactor licensees who decide to convert to Revision 4 of this guide. While Revision 4 is primarily intended for licensees of new nuclear power plants, the NRC staff sees no technical reason to prohibit a current operating reactor licensee from voluntarily making this conversion. Partial conversions (i.e., conversions only performed on particular variables or systems) are not recommended because of the potential for loss of variables or interactions with other variables without a complete analysis in accordance with this guide. If a current operating reactor licensee voluntarily converts to the criteria in Revision 4 of this guide, the licensee should perform the conversion on the plant's entire accident monitoring program.

3. Values and Impacts

In this regulatory analysis, the staff cannot quantify either the probability of guidance having a positive effect on accident monitoring or the probability of that effect on the achievement of overall safety goals. In the following summary of values and impacts, an impact represents a "cost" in terms of schedule, budget, staffing, or an undesired attribute that would accrue from taking the proposed approach.

3.1 *Alternative 1: Take no action*

This alternative has a perceived cost benefit, in that Revision 3 of this guide has been in use for more than 20 years and is familiar to the industry. However, its content is specific to the current fleet of light-water reactors. Moreover, it offers no guidance concerning the use of digital instrumentation. With advanced nuclear plant designs (unlike current light-water reactors) and digital instrumentation being used, many deviations from Revision 3 would be required.

Value: Industry familiarity with Revision 3 of Regulatory Guide 1.97
Impact: Schedule, budget, and staffing costs to the NRC and applicants, associated with evaluating deviations from Revision 3 of Regulatory Guide 1.97

3.2 *Alternative 2: Revise Regulatory Guide 1.97 to endorse IEEE Std. 497-2002 and incorporate deviations, clarifications, and rule changes that the NRC has previously approved for current operating reactor licensees*

This alternative would endorse IEEE Std. 497-2002 for both new plants and current operating reactors, and would carry forward and update Tables 1–3 and the associated discussion and regulatory positions from Revision 3 of the guide based on Branch Technical Position HICB-10 of the Standard Review Plan (NUREG-0800). The resultant guide would contain different sets of criteria for new plants (non-prescriptive) and current operating reactors (prescriptive), which could lead to ambiguities as to the plants (or class of plants) to which the criteria apply. Furthermore, the information carried forward and updated from Revision 3 would not provide any new guidance for current operating reactors and, hence, revising the guide would not yield any benefit.

Value:	Current criteria for both new plants and current operating reactors in Revision 4 of Regulatory Guide 1.97
Impact:	Possible ambiguities in applying criteria, and no new guidance for current operating reactors

3.3 *Alternative 3: Develop a new regulatory guide to endorse IEEE Std. 497-2002 for new plants, but leave Regulatory Guide 1.97 at Revision 3 for current operating reactors*

This alternative would require distinct guidance for current operating reactor licensees who use Revisions 2 or 3 of Regulatory Guide 1.97, which would differ from the new regulatory guide that endorses IEEE Std. 497-2002 for new plants. As such, this alternative would create an undesirable and potentially confusing situation for both licensees and regulators because the industry knows Regulatory Guide 1.97 as the sole source of accident monitoring criteria. This alternative would also require the NRC to revise each related regulatory document to reference both Regulatory Guide 1.97 and the new regulatory guide when referring to accident monitoring criteria. This would also create a temporary disconnect between the regulatory documents and the new regulatory guide for new plants until the NRC can revise all of these regulatory documents. This approach would be extremely labor- and time-intensive. As a result, the disadvantages and cost of implementing this approach far outweigh its limited added value.

Value:	Clear separation of criteria between new plants and current operating reactors
Impact:	Schedule, budget, and staffing costs to the NRC, associated with revising numerous regulatory documents to reference the new accident monitoring regulatory guide; accident monitoring criteria are known industry-wide to be in Regulatory Guide 1.97.

3.4 *Alternative 4: Revise Regulatory Guide 1.97 to endorse IEEE Std. 497-2002 for new plants and remove all previous guidance for current operating reactors*

This alternative is the most desirable approach because it would allow current operating reactors to use the existing methods to meet the NRC's accident monitoring regulations (based on Revision 2 or 3 of Regulatory Guide 1.97), while endorsing IEEE Std. 497-2002 (which reflects the current state of technology) for new plants. In addition, the performance-based criteria in the standard will minimize deviations and, thereby, minimize staff review costs. Moreover, since Revision 4 is intended for new plants, this approach would have no impact on current operating reactor licensees, unless the licensee voluntarily converts to its criteria.

Value:	Clear, current criteria for new plants, with no impact on current operating reactor licensees
Impact:	None, unless a current operating reactor licensee voluntarily converts to the criteria in Revision 4

4. Conclusion

The staff recommends that the NRC issue Revision 4 of Regulatory Guide 1.97 to endorse IEEE Std. 497-2002 for accident monitoring criteria. This action will enhance the licensing process for new nuclear power plants and improve the design and evaluation processes for accident monitoring instrumentation.

5. Backfit Analysis

This regulatory guide does not require a backfit analysis as described in 10 CFR 50.109(c) because it is intended for new nuclear power plants. The use of this revision by current operating reactor licensees is entirely voluntary.