U.S. NUCLEAR REGULATORY COMMISSION

DRAFT REGULATORY GUIDE DG-1335

Proposed Revision 5 to Regulatory Guide RG 1.97

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CRITERIA FOR ACCIDENT MONITORING INSTRUMENTATION FOR NUCLEAR POWER PLANTS

A. INTRODUCTION

Purpose

This regulatory guide (RG) describes an approach that is acceptable to the staff of the U.S. Nuclear Regulatory Commission (NRC) to meet regulatory requirements for instrumentation to monitor accidents in nuclear power plants. It endorses, with clarifications, the Institute of Electrical and Electronic Engineers (IEEE) Standard (Std.) 497-2016, "IEEE Standard Criteria for Accident Monitoring Instrumentation for Nuclear Power Generating Stations" (Ref. 1).

Applicability

This RG applies to all holders of operating licenses for nuclear power reactors under the provisions of Title 10 of the *Code of Federal Regulations* (10 CFR), Part 50, "Domestic Licensing of Production and Utilization Facilities" (Ref. 2), including those who have permanently ceased operations and have certified that fuel has been permanently removed from the reactor vessel. It also applies to certain holders of, and applicants for, a power reactor design certification or combined license under 10 CFR Part 52, "Licenses, Certifications, and Approvals for Nuclear Power Plants" (Ref. 3). Specifically, the method described in this RG relates to 10 CFR 50.34(f), "Additional TMI-Related Requirements," General Design Criterion (GDC) 13, "Instrumentation and Control," GDC 19, "Control Room," and GDC 64, "Monitoring Radioactivity Releases," as set forth in Appendix A, "General Design Criteria for Nuclear Power Plants," to 10 CFR Part 50, and certain design certification rules promulgated as appendices to 10 CFR Part 52.

Applicable Orders and Regulations

- NRC Order EA-12-051, "Order Modifying Licenses with Regard to Reliable Spent Fuel Pool Instrumentation," requires, in part, that all licensees have a reliable indication of the water level in associated spent fuel storage pools (Ref. 4).

This RG is being issued in draft form to involve the public in the development of regulatory guidance in this area. It has not received final staff review or approval and does not represent an NRC final staff position. Public comments are being solicited on this DG and its associated regulatory analysis. Comments should be accompanied by appropriate supporting data. Comments may be submitted through the Federal-rulemaking Web site, http://www.regulations.gov, by searching for draft regulatory guide DG-1335. Alternatively, comments may be submitted to the Rules, Announcements, and Directives Branch, Office of Administration, U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001. Comments must be submitted by the date indicated in the *Federal Register* notice.

Electronic copies of this DG, previous versions of this guide, and other recently issued guides are available through the NRC's public Web site under the Regulatory Guides document collection of the NRC Library at http://www.nrc.gov/reading-rm/doc-collections/reg-guides/. The DG is also available through the NRC's Agencywide Documents Access and Management System (ADAMS) at http://www.nrc.gov/reading-rm/adams.html, under Accession No. ML17083A134. The regulatory analysis may be found in ADAMS under Accession No. ML17083A133.

- 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities" provides regulations for licensing production and utilization facilities.

    - 10 CFR 50.34(f)(2)(xix) requires applicants and licensees to provide instrumentation adequate for monitoring plant conditions following an accident that includes core damage.

    - 10 CFR 50.55a, "Codes and Standards," describes consensus codes and standards that the NRC has incorporated into its regulations by reference.

    - 10 CFR Part 50, Appendix A, GDC 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.

    - 10 CFR Part 50, Appendix A, GDC 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. In addition, operating reactor licensees must provide equipment at appropriate locations outside the control room with a design capability for prompt hot shutdown of the reactor, including necessary instrumentation and controls to maintain the unit in a safe condition during hot shutdown.

    - 10 CFR Part 50, Appendix A, GDC 64, "Monitoring Radioactivity Releases," requires operating reactor licensees to provide the means for monitoring the reactor containment atmosphere, spaces containing components to recirculate loss-of-coolant accident fluids, effluent discharge paths, and the plant environs for radioactivity that may be released as a result of postulated accidents.

- 10 CFR Part 52, "Licenses, Certifications, and Approvals for Nuclear Power Plants," governs the issuance of early site permits, standard design certifications, combined licenses, standard design approvals, and manufacturing licenses for nuclear power facilities.

    - 10 CFR 52.47(a)(23), requires a design certification application for light-water reactor (LWR) designs to present a description and analysis of design features for the prevention and mitigation of severe accidents.

    - 10 CFR 52.79(a)(38), requires a combined license application for LWR designs to present a description and analysis of design features for the prevention and mitigation of severe accidents.

    - 10 CFR 52.137(a)(23), requires a standard design approval application for LWR designs to present a description and analysis of design features for the prevention and mitigation of severe accidents.

    - 10 CFR 52.157(f)(23), requires a manufacturing license application for LWR designs to present a description and analysis of design features for the prevention and mitigation of severe accidents.

    - The design certification document for each design is incorporated by reference in the associated appendix of 10 CFR Part 52. A certified design may include a requirement for the implementation of severe accident management guidelines (SAMGs). Consequently, for plants referencing such a certified design, the implementation of SAMGs is a regulatory requirement. For others, it is voluntary.

Related Guidance

- NUREG-0700, "Human-System Interface Design Reviews Guidelines" (Ref. 5), contains guidance to assist with the human factors engineering aspect of the nuclear power plant design review process and the interface between plant personnel and plant systems and components.

- NUREG-0711, "Human Factors Engineering Program Review Model" (Ref. 6), contains information to assist the NRC staff with the review of the human factors engineering programs.

- NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition" (Ref. 7).

    - NUREG-0800, Chapter 7, "Instrumentation and Controls," Branch Technical Position 7-19, "Guidance for Evaluation of Diversity and Defense-in-Depth in Digital Computer-Based Instrumentation and Control Systems," discusses the NRC's concerns about common cause failures and provides guidance for evaluating an applicant's diversity and defense-in-depth assessment appropriate.

    - NUREG-0800, Chapter 11, "Radioactive Waste Management," identifies guidance documents the NRC staff uses to determine whether the plant's design incorporates acceptable engineering practices and guidelines for systems containing radioactive material.

    - NUREG-0800, Chapter 18, "Human Factors Engineering," identifies guidance documents the NRC staff uses to determine whether the plant's design incorporates acceptable human factors engineering practices.

- RG 1.53, "Application of the Single-Failure Criterion to Nuclear Power Plant Protection Systems" (Ref. 8), endorses IEEE Std. 379, "IEEE Standard Application of the Single-Failure Criterion to Nuclear Power Generating Station Safety Systems" (Ref. 9), as an acceptable method to meet the regulations concerning the application of the single-failure criterion to the electrical power, instrumentation, and control portions of nuclear power plant safety systems.

- RG 1.152, "Criteria for Use of Computers in Safety Systems of Nuclear Power Plants" (Ref. 10), endorses IEEE Std. 7-4.3.2, "Standard Criteria for Digital Computers in Safety Systems of Nuclear Power Generating Stations" (Ref. 11), as an acceptable method to meet the regulations. IEEE 7-4.3.2 provides criteria supporting the specification, design, and implementation of computers in safety systems of nuclear power generating stations. It should be used in conjunction with the he most recent version of IEEE Std. 603, "IEEE Standard Criteria for Safety Systems for Nuclear Power Generating Stations" (Ref. 12), that is incorporated by reference in 10 CFR 50.55a. to ensure completeness of the post-accident monitoring system design when a computer is to be used as a component of a post-accident monitoring system to enable the ability of plant operators to take preplanned manual safety actions for which there are no provisions for automatic safety actions.

Paperwork Reduction Act

This RG provides guidance for implementing the mandatory information collections in 10 CFR Parts 50 and 52 that are subject to the Paperwork Reduction Act of 1995 (44 U.S.C. 3501 et. seq.). These information collections were approved by the Office of Management and Budget (OMB), under control numbers 3150-0011 and 3150-0151. Send comments regarding this information collection to the Information Services Branch, U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001, or by e-mail to Infocollects.Resource@nrc.gov, and to the Desk Officer, Office of Information and Regulatory Affairs NEOB-10202, (3150-0011, 3150-0151), Office of Management and Budget, Washington, DC 20503.

Public Protection Notification

The NRC may not conduct or sponsor, and a person is not required to respond to, a collection of information unless the document requesting or requiring the collection displays a currently valid OMB control number.

B. DISCUSSION

Reason for Revision

The staff is issuing Revision 5 of RG 1.97 to endorse IEEE Std. 497-2016 "Criteria for Accident Monitoring Instrumentation for Nuclear Power Plants," with exceptions and clarifications. Revision 5 also makes further clarifying revisions by expressly expanding the applicability of RG 1.97 to holders of, or applicants for, power reactor design certifications or combined licenses under 10 CFR Part 52, and by adding references to the NRC's 10 CFR Part 52 regulations and related NRC guidance documents.

Background

Revision 4 of RG 1.97, endorses IEEE Std. 497-2002, "IEEE Standard Criteria for Accident Monitoring Instrumentation for Nuclear Power Generating Stations" (Ref. 13). Since the release of Revision 4 of RG 1.97, the IEEE revised Std. 497 in 2010 and again in 2016 to reflect a more technology-neutral approach and to bring the IEEE standard more in line with the international standards referenced in the IEEE Std. 497-2016 and the Harmonization section below. In addition to the Types A, B, C, D, and E variables defined in the previous revisions, IEEE Std. 497-2016 adds a new Type F variable, which provides primary information to indicate fuel damage and the effects of fuel damage.

In March 1979, an accident occurred in Unit 2 of the Three Mile Island Nuclear Station. In the aftermath, the nuclear industry and the NRC adopted a more rigorous approach to accident monitoring. In May 1983, the NRC issued Revision 3 of RG 1.97, "Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant and Environs Conditions During and Following an Accident" (Ref. 14). The RG prescribed a detailed list of variables to monitor, and specified a comprehensive list of design and qualification criteria to be met.

Because of its prescriptive nature, RG 1.97 quickly became the de facto standard for accident monitoring. With the increased use of digital instrumentation systems in advanced nuclear power plant designs, the nuclear industry recognized a need to develop a consolidated standard that was more flexible. Instead of prescribing the instrument variables to be monitored (as was the case in Revision 3 of RG 1.97), the industry developed performance-based criteria for use in selecting variables. These efforts resulted in the development of IEEE Std. 497-2002, which established flexible, performance-based criteria for the selection, performance, design, qualification, display, and quality assurance of accident monitoring variables, which are the operators' primary sources of accident monitoring information. This standard was endorsed by Revision 4 of RG 1.97, subject to eight regulatory positions. IEEE Std. 497-2010 (Ref. 15) was developed to incorporate some of those regulatory positions and revised some definitions and terminology.

Based on insights from the Fukushima Dai-ichi accident in March 2011, the nuclear industry in the United States has recognized the need for instrumentation to monitor plant conditions after fuel damage has occurred. In addition, the NRC has determined that all power reactor licensees must have a reliable means of remotely monitoring wide-range spent fuel pool levels. As a result, the NRC issued an order (NRC Order EA-12-051) to all licensees requiring spent fuel pool level instrumentation.

Accordingly, IEE broadened the scope of the standard to include severe accidents. IEEE Std. 497-2016 reflects a more technology-neutral approach and an effort to bring the standard in line with related international standards.

In addition, IEEE Std. 497-2016 adds a new kind of variable, Type F, which provides primary information to indicate fuel damage and the effects of fuel damage. The regulatory requirement for Type F variables derives from specific appendices of 10 CFR Part 52. At least one of the design certification rules incorporates, by reference, a design certification document that requires combined license applicants to implement SAMGs. Because the SAMGs cannot be implemented without instrumentation to determine what actions are needed during a severe accident, these licensees require Type F variables. For other plants, implementation of SAMGs is voluntary. However, as of the date of issue of this RG, every nuclear power plant in the United States has voluntarily implemented SAMGs. The plants cannot do so without an equivalent to Type F variables, and therefore this guidance may be useful to them. IEEE Std. 497-2016 also includes four informative annexes:

- Annex A, "Accident monitoring instrument channel accuracy," provides general guidance. Clause A.2 provides guidance on accuracy requirement groupings according to the control room personnel usage. Clause A.3 provides typical accuracy requirements.

- Annex B, "Examples of monitoring channel displays," provides examples, as figures, of how sensors, signal processing, data validation, and displays may be configured to provide accident monitoring instrumentation.

- Annex C, "Operational state diagram," provides a table showing the relationship of the plant state to the procedure used for the various plant states (for all states from normal operation to severe accidents).

- Annex D, "Bibliography," lists the references cited in the standard and identifies how they may be obtained.

Harmonization with International Standards

The NRC has a goal of harmonizing its guidance with international standards, to the extent practical. The International Atomic Energy Agency (IAEA) have issued a significant number of standards, guidance and technical documents, and recommendations addressing good practices in most aspects of radiation protection, including:

- IAEA Nuclear Energy Series No. NP-T-3.16, "Accident Monitoring Systems for Nuclear Power Plants," February 2015 (Ref. 16).

- IAEA Safety Standard Series No. TECDOC-1818, "Assessment of Equipment Capability to Perform Reliably under Severe Accident Conditions," July 2017 (Ref. 17).

- IAEA Safety Standards Series No. SSG-39, "Design of Instrumentation and Control Systems for Nuclear Power Plants," April 2016 (Ref. 18).

This RG incorporates similar design and performance guidelines and is consistent with the safety principles provided in these IAEA publications.

Documents Discussed in Staff Regulatory Guidance

This RG endorses the use of one or more codes or standards developed by external organizations and other third party guidance documents. These codes, standards, and third party guidance documents may contain references to other codes, standards or third party guidance documents ("secondary references”). If a secondary reference is incorporated by reference into NRC regulations as a requirement, then licensees and applicants must comply with that standard as set forth in the regulation. If the secondary reference is endorsed in a RG as an acceptable approach for meeting an NRC requirement, then the standard constitutes a method acceptable to the NRC staff for meeting that regulatory requirement as described in the specific RG. If the secondary reference is neither incorporated by reference into NRC regulations nor endorsed in a RG, then the secondary reference is neither a legally-binding requirement nor a “generic” NRC approved acceptable approach for meeting an NRC requirement. However, licensees and applicants may consider and use the information in the secondary reference, if appropriately justified and consistent with applicable NRC requirements and current regulatory practice.

C. STAFF REGULATORY GUIDANCE

This RG endorses IEEE Std. 497-2016 as an acceptable method for providing instrumentation to monitor variables for accident conditions, subject to the following regulatory guidance positions:

1. The applicability of IEEE Std. 497-2016 for licensees and applicants already committed to an earlier revision of RG 1.97 should be clarified:

   A licensee or applicant committed to an earlier revision of RG 1.97 may voluntarily convert the entire accident monitoring program using the criteria in this revision. To do so, the licensee or applicant needs to analyze the accident monitoring program in its entirety.

   Having carefully considered the applicability and usefulness of the new standard, the NRC staff recognizes that licensees or applicants may be interested in converting to the latest revision of RG 1.97, updating the plant's entire accident monitoring program from the current licensing basis to a less prescriptive one. This conversion would entail conducting a new analysis of every type of variable, and may result in physical modifications to the plant and changes to its procedures and technical specifications.

2. Licensees and applicants that have committed to an earlier revision of RG 1.97 may voluntarily add Type F variables, that is, those variables to be monitored while managing a severe accident:

   A licensee or applicant committed to an earlier revision of RG 1.97 may use this revision to add Type F variables to their plant's accident monitoring program. The licensee or applicant should first perform a comprehensive analysis of severe accidents to determine the variables to be selected.

   This revision of RG 1.97 may be used to select only Type F variables and determine their design, performance, qualification, and display criteria. For these modifications, the licensee should perform an analysis addressing only the severe accident variables in accordance with the selection criteria in this revision. When a licensee elects to adopt this revision for only Type F variables and remains committed to an earlier version of RG 1.97 for variables of Type A, B, C, D, and E, the licensee should document how it applied each revision of RG 1.97.

3. The scope of variables analyzed as Type A should include variables that are associated with contingency actions. Specifically, the application of the term "contingency actions" should be modified by deleting the words "**do not**" from the next-to-last sentence in Clause 4.1, "Type A variables." The revised sentence is clarified 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.*

   IEEE Std. 497-2016 defines "contingency actions" as "alternative actions taken to address unexpected responses of the plant or conditions beyond its licensing basis (e.g., actions taken for multiple equipment failures)." Clause 4.1 of IEEE 497-2016 uses this term in defining selection criteria for Type A variables. The NRC staff agrees with the criteria in this clause, except for the exclusion of variables that are associated with contingency actions. Nuclear steam supply system vendors have not used this term consistently in emergency procedure guidelines. Therefore, the NRC staff does not endorse the exclusion of variables associated with contingency actions from the selection criteria (especially those associated with plant-specific operating procedures or guidelines). Rather, the scope of the analysis for variables of this type should be as inclusive as possible. Thus contingency variables may or may not be selected as a result of that analysis.

4. The ranges of instrumentation should include appropriate margins, as noted by the following clarifications to Clause 5.1, "Range":

   In the second paragraph, the second sentence is clarified as follows:

   > *These variables shall have extended ranges, sufficient range to cover, with appropriate margin, the predicted limits of the variables and address a source term that considers a fuel damage.*

   The second paragraph is clarified to include the following:

   > *The range of instrumentation used to implement EOPs* [emergency operating procedures] *should cover, with appropriate margin, the predicted full range of the variables with the consideration of analytical uncertainties and environment measurement errors under design-basis accident conditions.*

   The last paragraph, is clarified to include the following:

   > *The instrumentation used to implement SAMGs* [severe accident monitoring guidelines] *should have sufficient range to cover, with appropriate margin, the predicted limits of the variables.*

5. The accuracy of instrumentation should be derived from the licensing basis, and is clarified to include the following to Clause 5.2, "Accuracy":

   > *Instrumentation used to implement EOPs must fulfill accuracy requirements derived from the plant design basis. During SAMG implementation, although the value of a variable may be needed, it is usually more important to follow the trends of the variables used. The accuracy requirements specified for each severe accident mitigation strategy are based on the level of accuracy needed for decision making for support of the mitigation strategy or for support of emergency preparedness functions. If redundant instrumentation is provided, or where spatial orientation and distribution is required for measuring the same parameter, the equipment must be sufficiently accurate to preclude ambiguous trend information.*

   In addition, IEEE Std. 497-2016, Annex A, 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% of full span." The licensee should base the accuracy of the instrument channels on the plant's licensing basis or the severe accident analysis and not the example provided in this section.

6. *IEEE Std. 497-2016 does not mention the number of measurement points for each variable (with the exception of redundancy requirements).* The number of measurement points for each variable should be sufficient to adequately characterize the variable.

For example, containment temperature may require several measurement points, spatially distributed.

7. The design of instrumentation should incorporate diversity and defense-in-depth as part of addressing common cause failures.

   Clause 6.2, "Common cause failures" in IEEE Std. 497-2016 states:

   > *Design of Type A, Type B, and Type C instrumentation shall address common cause failures, as described in IEEE Std. 379-2016 and IEEE Std. 603-2009/ IEC 62340:2007 consistent with the plant's LBD* [licensing basis document]. *For instrumentation using digital devices, guidance to address common cause failures can be found in IEEE Std. 7-4.3.2-2016/IEC 60880:2006.*

   The licensee or applicant should verify the acceptability of endorsed IEEE or International Electrotechnical Commission (IEC) standards identified in Clause 6.2 of IEEE Std. 497-2016; alternately, applicants should consider NUREG-0800, Chapter 7, Branch Technical Position 7-19, "Guidance for Evaluation of Diversity and Defense-in-Depth in Digital Computer-Based Instrumentation and Control Systems." Branch Technical Position 7-19 describes the NRC staff's concerns about common cause failures and provides guidance for evaluating an applicant's diversity and defense-in-depth assessment, design, and the design of manual controls and displays to ensure conformance with the NRC position on diversity and defense-in-depth for instrumentation and control systems incorporating digital, software-based or software-logic-based reactor trip systems, engineered safety features, auxiliary supporting features, and other auxiliary features as appropriate.

   - The use of secondary references is clarified in this RG above in the Section B subsection titled, "Documents Discussed in Staff Regulatory Guidance." The following secondary references should be reviewed to determine the applicable version or revision:

     - IEEE Std. 379, "IEEE Standard for Application of the Single-Failure Criterion to Nuclear Power Generating Station Safety Systems," is discussed in RG 1.53.

     - IEEE Std. 603, "IEEE Standard Criteria for Safety Systems for Nuclear Power Generating Stations," is incorporated by reference in 10 CFR 50.55a(h) "Protection and Safety Systems."

     - IEEE Std. 7-4.3.2, "IEEE Standard Criteria for Programmable Digital Devices in Safety Systems of Nuclear Power Generating Stations," is discussed in RG 1.152.

8. The design of instrumentation should incorporate human factors guidelines or accepted human factor methods or principles.

   Clauses 5.3, "Response time," 8.1.2, "Human factors," and 8.5, "Display location," all identify human factor concerns. However, IEEE Std. 497-2016 does not identify any specific human factors guidelines or accepted human factors methods or principles. Consequently, applicants should consider Chapter 18, "Human Factors Engineering," of NUREG-0800, which identifies guidance documents the NRC staff uses to determine if acceptable human factors engineering practices and guidelines have been incorporated into a plant's design. These guidance documents include NUREG-0711, and NUREG-0700. Additionally, NUREG-0711 states that a practice, method, or guide is "accepted" if it is (1) documented in the human factors literature within a standard or guidance document that underwent a peer-review process, or (2) is justified through scientific research or industrial practices.

9. The following corrections are addressed for clarity:

a) IEEE Std. 497-2016, Section 4.7, the description of a Type F variable under the second column "Selection criteria for the variable type" of Table 1 "Summary of accident monitoring variable types/source documents" states:

> *Monitor the direct effects (e.g. combustible gases concentration, radiation, pressure, or temperature of fuel damage).*

The wording "temperature of fuel damage" can cause confusion and in is clarified as follows:

> *Monitor the direct effects of fuel damage (e.g. combustible gases concentration, radiation, pressure, or temperature).*

b) IEEE Std. 497-2016, Section 3, "Definitions," revises the definition of anticipated operational occurrence (AOO). This term was already defined in 10 CFR Part 50, Appendix A, and the definition in previous versions of IEEE Std 497 was identical to the definition in 10 CFR Part 50, Appendix A. While it does not appear that the intent was to significantly change in the way the standard uses the term, the NRC only endorses the 10 CFR Part 50, Appendix A definition.

D. IMPLEMENTATION

The purpose of this section is to provide information on how applicants and licensees[1] may use this RG and information regarding the NRC's plans for using this RG. In addition, it describes how the NRC staff complies with 10 CFR 50.109, "Backfitting," and any applicable issue finality provisions in 10 CFR Part 52, "Licenses, Certifications, and Approvals for Nuclear Power Plants."

Use by Applicants and Licensees

Applicants and licensees may voluntarily[2] use the guidance in this document to demonstrate compliance with the underlying NRC regulations. Methods or solutions that differ from those described in this RG may be deemed acceptable if they provide sufficient basis and information for the NRC staff to verify that the proposed alternative demonstrates compliance with the appropriate NRC regulations. Current licensees may continue to use guidance the NRC found acceptable for complying with the identified regulations as long as their current licensing basis remains unchanged.

Licensees may use the information in this RG for actions that do not require NRC review and approval such as changes to a facility design under 10 CFR 50.59, "Changes, Tests, and Experiments." Licensees may use the information in this RG or applicable parts to resolve regulatory or inspection issues.

Use by NRC Staff

The NRC staff does not intend or approve any imposition or backfitting of the guidance in this RG. The NRC staff does not expect any existing licensee to use or commit to using the guidance in this RG, unless the licensee makes a change to its licensing basis. The NRC staff does not expect or plan to request licensees to voluntarily adopt this RG to resolve a generic regulatory issue. The NRC staff does not expect or plan to initiate NRC regulatory action which would require the use of this RG. Examples of such unplanned NRC regulatory actions include issuance of an order requiring the use of the RG, requests for information under 10 CFR 50.54(f) as to whether a licensee intends to commit to use of this RG, generic communication, or promulgation of a rule requiring the use of this RG without further backfit consideration.

During regulatory discussions on plant-specific operational issues, the staff may discuss with licensees various actions consistent with staff positions in this RG, as one acceptable means of meeting the underlying NRC regulatory requirement. Such discussions would not ordinarily be considered backfitting even if prior versions of RG 1.97 are part of the licensing basis of the facility. However, unless this RG is part of the licensing basis for a facility, the staff may not represent to the licensee that the licensee's failure to comply with the positions in this RG constitutes a violation of underlying NRC regulatory requirements.

If an existing licensee voluntarily seeks a license amendment or change and (1) the NRC staff's consideration of the request involves a regulatory issue directly relevant to this RG and (2) the specific

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1 In this section, "licensees" refers to licensees of nuclear power plants under 10 CFR Parts 50 and 52; and "applicants," refers to applicants for licenses and permits for (or relating to) nuclear power plants under 10 CFR Parts 50 and 52, and applicants for standard design approvals and standard design certifications under 10 CFR Part 52.

2 In this section, "voluntary" and "voluntarily" mean that the licensee is seeking the action of its own accord, without the force of a legally binding requirement or an NRC representation of further licensing or enforcement action.

subject matter of this RG is an essential consideration in the staff's determination of the acceptability of the licensee's request, then the staff may request that the licensee either follow the guidance in this RG or otherwise demonstrate compliance with the underlying NRC regulatory requirements. This is not considered backfitting as defined in 10 CFR 50.109(a)(1). Neither is it a violation of any of the issue finality provisions in 10 CFR Part 52.

Additionally, an existing applicant may be required to comply with new rules, orders, or guidance if 10 CFR 50.109(a)(3) applies.

If a licensee believes that the NRC is either using this RG or requesting or requiring the licensee to implement the methods or processes in this RG in a manner inconsistent with the discussion in this Implementation section, then the licensee may file a backfit appeal with the NRC in accordance with the guidance in NRC Management Directive 8.4, "Management of Facility-Specific Backfitting and Information Collection" (Ref. 19), and NUREG-1409, "Backfit Guidelines" (Ref. 20).

REFERENCES [3]

1. Institute of Electrical and Electronic Engineers, (IEEE) Standard (Std.) 497-2016, "IEEE Standard Criteria for Accident Monitoring Instrumentation for Nuclear Power Generating Stations," 2016, New York, NY.[4]

2. *U.S. Code of Federal Regulations* (CFR), "Domestic Licensing of Production and Utilization Facilities," Part 50, Chapter 1, Title 10, "Energy."

3. CFR, "Licenses, Certifications, and Approvals of Nuclear Power Plants," Part 52, Chapter 1, Title 10, "Energy."

4. U.S. Nuclear Regulatory Commission (NRC), Order EA-12-051, "Order Modifying Licenses with Regard to Spent Fuel Pool Instrumentation (Effective Immediately)," March 12, 2012, Washington, DC.

5. NRC, NUREG-0700, "Human-System Interface Design Review Guideline: Review Methodology and Procedures," Washington DC.

6. NRC, NUREG-0711, "Human Factors Engineering Program Review Model," Washington DC.

7. NRC, NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition,"" Washington DC.

8. NRC, Regulatory Guide (RG) 1.53, "Application of the Single-Failure Criterion to Nuclear Power Plant Protection Systems," Washington DC.

9. IEEE, Std. 379, "IEEE Standard Application of the Single-Failure Criterion to Nuclear Power Generating Station Safety Systems," New York, NY.

10. NRC, RG 1.152, "Criteria for Use of Computers in Safety Systems of Nuclear Power Plants," Washington DC.

11. IEEE, Std. 7-4.3.2, "IEEE Standard Criteria for Digital Computers in Safety Systems of Nuclear Power Generating Stations," 1993, New York, NY

12. IEEE, Std. 603-2009, "IEEE Standard Criteria for Safety Systems for Nuclear Power Generating Stations" November 2009, New York, NY.

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3 Publicly available NRC published documents are available electronically through the NRC Library on the NRC's public Web site at http://www.nrc.gov/reading-rm/doc-collections/ and through the NRC's Agencywide Documents Access and Management System (ADAMS) at http://www.nrc.gov/reading-rm/adams.html. The documents can also be viewed online or printed for a fee in the NRC's Public Document Room (PDR) at 11555 Rockville Pike, Rockville, MD. For problems with ADAMS, contact the PDR staff at 301-415-4737 or (800) 397-4209; fax (301) 415-3548; or e-mail pdr.resource@nrc.gov.

4 Copies of Institute of Electrical and Electronics Engineers (IEEE) documents may be purchased from the Institute of Electrical and Electronics Engineers Service Center, 445 Hoes Lane, PO Box 1331, Piscataway, NJ 08855 or through the IEEE's public Web site at http://www.ieee.org/publications_standards/index.html.

13. NRC, RG 1.97, Revision 4, "Criteria for Accident Monitoring Instrumentation for Nuclear Power Plants," June 2006, Washington DC.

14. NRC, RG 1.97, Revision 3, "Criteria for Accident Monitoring Instrumentation for Nuclear Power Plants," May 1983, Washington DC.

15. IEEE Std. 497-2010, "IEEE Standard Criteria for Accident Monitoring Instrumentation for Nuclear Power Generating Stations," November 2010, New York, NY.

16. International Atomic Energy Agency (IAEA), IAEA Nuclear Energy Series No. NP-T-3.16, "Accident Monitoring Systems for Nuclear Power Plants," February 2015, Vienna, Austria.[5]

17. IAEA, IAEA Safety Standards Series No. TECDOC-1818, "Assessment of Equipment Capability to Perform Reliably under Severe Accident Conditions," July 2017, Vienna, Austria.

18. IAEA, IAEA Safety Standards Series No. SSG-39, "Design of Instrumentation and Control Systems for Nuclear Power Plants," April 2016, Vienna, Austria.

19. NRC Management Directive 8.4, "Management of Facility-Specific Backfitting and Information Collection," October 2013, Washington DC.

20. NRC, NUREG-1409, "Backfitting Guidelines," Washington DC.

21. IEC, Std. 60880:2006, "Nuclear Power Plants – Instrumentation and Control Systems Important to Safety – Software Aspects for Computer-Based Systems Performing Category A Functions," 2006, Geneva, Switzerland.

5 Copies of International Atomic Energy Agency (IAEA) documents may be obtained through their Web site: www.IAEA.Org or by writing the International Atomic Energy Agency P.O. Box 100 Wagramer Strasse 5, A-1400 Vienna, Austria. Telephone (+431) 2600-0, Fax (+431) 2600-7, or E-Mail at Official.Mail@IAEA.Org.