NUREG-0800

U.S. NUCLEAR REGULATORY COMMISSION

STANDARD REVIEW PLAN

BRANCH TECHNICAL POSITION 7-12

GUIDANCE ON ESTABLISHING AND MAINTAINING INSTRUMENT SETPOINTS

REVIEW RESPONSIBILITIES

Primary – Organization responsible for the review of instrumentation and controls

Secondary – None

Review Note: The revision numbers of Regulatory Guides (RG) and the years of endorsed industry standards referenced in this branch technical position (BTP) are centrally maintained in Standard Review Plan (SRP) Section 7.1-T (Table 7-1). Therefore, the individual revision numbers of RGs (except RG 1.97) and years of endorsed industry standards are not shown in this BTP. References to industry standards incorporated by reference into regulation (IEEE Std 279-1971 and IEEE Std 603-1991) and industry standards that are not endorsed by the agency do include the associated year in this BTP. See Table 7-1 to ensure that the appropriate RGs and endorsed industry standards are used for the review.

Draft Revision 6 – August 2015

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A. BACKGROUND

This branch technical position (BTP) provides guidelines for reviewing the process an applicant~~/~~ or licensee follows to establish and maintain instrument setpoints. These guidelines are based on reviews of applicant~~/~~ or licensee submittals and vendor topical submittals describing setpoint assumptions, terminology, and methodology, and on experience gained from U.S. Nuclear Regulatory Commission (NRC) inspections of operating plants.

1. Regulatory Basis

Title 10 of the *Code of Federal Regulations* (10 CFR) 50.55a(h), ~~"~~“Protection and Safety Systems~~,"~~,” requires compliance with Institute of Electrical and Electronics Engineers (IEEE
) Standard (Std~~.~~) 603-1991, ~~"~~“IEEE Standard Criteria for Safety Systems for Nuclear Power Generating Stations~~,"~~,” and the correction sheet dated January 30, 1995. ~~–~~For nuclear plants with construction permits issued before January 1, 1971, the applicant~~/~~ or licensee may elect to comply instead with the plant-specific licensing basis. For nuclear power plants with construction

~~Revision 5 - March 2007~~

~~USNRC STANDARD REVIEW PLAN~~

~~This Standard Review Plan, NUREG-0800, has been prepared to establish criteria that the U.S. Nuclear Regulatory Commission staff responsible for the review of applications to construct and operate nuclear power plants intends to use in evaluating whether an applicant/licensee meets the NRC's regulations. The Standard Review Plan is not a substitute for the NRC's regulations, and compliance with it is not required. However, an applicant is required to identify differences between the design features, analytical techniques, and procedural measures proposed for its facility and the SRP acceptance criteria and evaluate how the proposed alternatives to the SRP acceptance criteria provide an acceptable method of complying with the NRC regulations.~~

~~The standard review plan sections are numbered in accordance with corresponding sections in Regulatory Guide 1.70, "Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants (LWR Edition)." Not all sections of Regulatory Guide 1.70 have a corresponding review plan section. The SRP sections applicable to a combined license application for a new light-water reactor (LWR) are based on Regulatory Guide 1.206, "Combined License Applications for Nuclear Power Plants (LWR Edition)."~~

~~These documents are made available to the public as part of the NRC's policy to inform the nuclear industry and the general public of regulatory procedures and policies. Individual sections of NUREG-0800 will be revised periodically, as appropriate, to accommodate comments and to reflect new information and experience. Comments may be submitted electronically by email to NRR_SRP@nrc.gov.~~

~~Requests for single copies of SRP sections (which may be reproduced) should be made to the U.S. Nuclear Regulatory Commission, Washington, DC 20555, Attention: Reproduction and Distribution Services Section, or by fax to (301) 415-2289; or by email to DISTRIBUTION@nrc.gov. Electronic copies of this section are available through the NRC's public Web site at http://www.nrc.gov/reading-rm/doc-collections/nuregs/staff/sr0800/, or in the NRC's Agencywide Documents Access and Management System (ADAMS), at http://www.nrc.gov/reading-rm/adams.html, under Accession # ML070550078.~~

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permits issued between January 1, 1971, and May 13, 1999, the applicant~~/~~ or licensee may elect to comply with the requirements stated in IEEE Std~~.~~ 279-1971, ~~"~~“Criteria for Protection Systems for Nuclear Power Generating Stations~~."~~.” Clause 4.4 of IEEE Std~~.~~ 603-1991 requires identification of the analytical limit associated with each variable. Clause 6.8.1 requires that allowances for uncertainties between the analytical limit and the device setpoint be determined using a documented methodology. Clause 3(6) of IEEE Std 279-1971 requires identification of the levels that, when reached, will necessitate protective action.

10 CFR Part 50, “Domestic Licensing of Production and Utilization Facilities,” Appendix B, “Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants,” Criterion

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XI, “Test Control,” and Criterion XII, ~~""~~“Control of Measuring and Test Equipment~~,"~~,” provide requirements for tests and test equipment used in maintaining instrument setpoints.

10 CFR Part 50, Appendix A, “General Design Criteria for Nuclear Power Plants,” General Design Criterion (GDC) 13, “Instrumentation and Control,” requires, in part, that instrumentation be provided to monitor variables and systems~~,~~ and that controls be provided to maintain these variables and systems within prescribed operating ranges.

GDC 20, “Protection System Functions,” requires, in part, that the protection system be designed to initiate automatically the operation of appropriate systems including the reactivity control systems, to assure that specified acceptable fuel design limits are not exceeded as a result of anticipated operational occurrences.

10 CFR 50.36(c)(1)(ii)(A), “Technical Specifications,” requires, in part that, where a limiting safety system setting (LSSS) is specified for a variable on which a safety limit has been placed, the setting be so chosen that automatic protective action will correct the abnormal situation before a safety level is exceeded. LSSSs are settings for automatic protective devices related to variables with significant safety functions. Setpoints found to exceed technical specification limits are considered as malfunctions of an automatic safety system. Such an occurrence could challenge the integrity of the reactor core, reactor coolant pressure boundary, containment, and associated systems.

10 CFR 50.36(c)(3), “Technical Specifications,” states that surveillance requirements are requirements relating to test, calibration, or inspection to assure that the necessary quality of systems and components is maintained, that facility operation will be within safety limits, and that the limiting conditions for operation will be met.

2. Relevant Guidance

~~Regulatory Guide~~

RG 1.105~~, Revision 3~~, “Setpoints for Safety-Related Instrumentation,” provides guidance for ensuring that instrument setpoints are initially ~~--~~ and remain ~~--~~ within the technical specification limits. This ~~regulatory guide~~RG endorses International Society of Automation (ISA-)~~-~~S67.04~~-1994~~, Part I, “Setpoints for Nuclear Safety-Related Instrumentation Used in Nuclear Power Plants.”

ISA-S67.04-1994, Part II, “Methodology for the Determination of Setpoints for Nuclear Safety-Related Instrumentation,” provides additional guidance, but ~~Regulatory Guide~~RG 1.105~~, Revision 3,~~ does not endorse or address Part II of ISA-S67.04-1994.

NRC Regulatory Issue Summary (RIS) 2006-17, “NRC Staff Position on the Requirements of 10 CFR 50.36, ‘Technical Specifications,~~'~~’ Regarding Limiting Safety System Settings During Periodic Testing and Calibration of Instrument Channels,” discusses issues that could occur during testing of LSSS.

IEEE Std~~.~~ 498-1990, ~~""~~“IEEE Standard Requirements for the Calibration and Control of

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Measuring and Test Equipment Used in Nuclear Facilities~~,"~~," and American National Standards Institute (ANSI~~/~~)/NCSL ~~Std.~~ Z540-1-1994, ~~"~~"Calibration Laboratories and Measuring and Test Equipment - General Requirements~~,"~~," provide guidance for the calibration and control of measuring and test equipment used in the maintenance of instrument setpoints.

Generic Letter (GL) 91-04, ~~"Guidance on Preparation of a Licensee Amendment Request for~~ "Changes in Technical Specification Surveillance Intervals to Accommodate a 24-Month Fuel Cycle~~,"~~," provides guidance on issues that should be addressed by the setpoint analysis when calibration intervals are extended from 12 or 18 to 24 months.
~~Appendix 7.1-C provides Standard Review Plan (~~
SRP~~) acceptance criteria for safety system compliance with 10 CFR 50.55a(h).~~
Appendix 7.1-B provides ~~SRP acceptance criteria for protection system compliance with 10 CFR 50.55a(h).~~guidance for evaluating conformance to the requirements of IEEE Std 279-1971.

SRP Appendix 7.1-C provides guidance for evaluating conformance to IEEE Std 603-1991.

SRP Appendix 7.1-D provides ~~SRP~~guidance for evaluating conformance to the acceptance criteria ~~for digital I&C compliance with~~contained in RG 1.152, "Criteria for Use of Computers in Safety Systems of Nuclear Power Plants," which endorses IEEE Std~~.~~ 7-4.3.2~~-2003, "~~, "IEEE Standard Criteria for Digital Computers in Safety Systems of Nuclear

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Power Generating Stations~~," as endorsed by Regulatory Guide 1.152, Revision 2, "Criteria for Digital Computers in Safety Systems of Nuclear Power Plants."~~."

~~Regulatory Guide~~
RG 1.174, ~~Revision 1, "~~"An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis~~,"~~," provides guidance on the use of probabilistic risk assessment (PRA) findings and risk insights in support of licensee requests for changes to a ~~plant's~~plant's licensing basis, as in requests for licensing amendments and technical specification changes.

~~Regulatory Guide~~
RG 1.177, ~~"~~"An Approach for Plant-Specific, Risk-Informed Decision Making: Technical Specifications~~,"~~," provides guidance on assessing the nature and impact of proposed technical specification changes by considering engineering issues and applying risk insights.

~~Regulatory Guide~~
RG 1.200 (For Trial Use), ~~"~~"An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities~~,"~~," provides guidance on determining that the quality of the PRA, in total or the parts that are used to support an application, is sufficient to provide confidence in the results such that the PRA can be used in regulatory ~~decision making~~decisionmaking for light-water reactors.

3. Definitions

ISA-S67.04~~-1994~~, Part I, Section 3 and Figure 1 provide acceptable definitions (except as noted by ~~Regulatory Guide~~RG 1.105~~, Revision 3,~~ and RIS-2006-17) of setpoint terminology and relationships between trip setpoint, allowable value, analytical limit, LSSS, and safety limit. The following additional definitions are provided for reviewer guidance:

~~A.~~ A. Acceptable as-found band: It is the band around the nominal trip setpoint or previous as-left setting of the instrument within which the as-found setpoint is expected to fall. The band accounts for the uncertainties associated with factors such as instrument reference accuracy, measurement and test equipment (MT&E), readability, normal environment effect, and drift of the instrument components that are being tested, and it accounts only for the duration between the tests. The width of the band is established by the ~~Deviation Limit~~deviation limit (DL), which may be asymmetrical relative to the reference value (nominal setpoint (NSP) or previous as-left) and defines the deviation (from the previous as-left value or NSP) that is expected to occur during the test. It should be noted that the DL must not include the setting tolerance (ST).

~~B.~~ B. As-left tolerance band or acceptable as-left band: It is the band around the nominal trip setpoint (LSP) - or around any value which is more conservative than the LSP - within which the as-left setpoint must fall at the conclusion of a channel test. The band accounts for the as-left tolerance, which some licensees define as leeway given to the instrument technician or calibration tolerance or setting tolerance. Setting tolerance can be based upon particular uncertainties such as reference accuracy, MT&E, and readability, but the total loop uncertainty analysis must explicitly account for each of these uncertainty terms whether or not the ST incorporates these uncertainties. ST may also be a specified value selected on

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the basis of engineering judgment or other consideration. However, in that situation, the as-found value must be compared with the previous as-left value.

4. Purpose

The purpose of this BTP is to provide guidance for the NRC staff to verify conformance with the previously cited regulatory bases and standards for instrument setpoints. This BTP has three objectives:

~~To verify~~

- Verify that setpoint calculation methods are adequate to ~~assure~~ensure that protective actions are initiated before the associated plant process parameters exceed their analytical limits.

~~To verify~~

- Verify that setpoint calculation methods are adequate to ~~assure~~ensure that control and monitoring setpoints are consistent with their requirements.

~~To confirm~~

- Confirm that the established calibration intervals and methods are consistent with safety analysis assumptions.

B. BRANCH TECHNICAL POSITION

1. Introduction

Instrumentation and control (I&C) safety systems control plant parameters to ~~assure~~ensure that safety limits will not be exceeded under the most severe design~~-~~basis accident. Instrument setpoints and acceptable as-left and acceptable as-found bands for these I&C safety system functions are chosen so that potentially unsafe or damaging process excursions (transients) can be avoided and/or terminated before plant conditions exceed safety limits. Accident analyses establish the limits for critical process parameters. These analytical limits, as established by accident analyses, do not normally include considerations for the accuracy (uncertainty) of installed instrumentation. Additional analyses and procedures are necessary to ~~assure~~ensure that the limiting trip setpoint of each safety control function is appropriate.

Instrument channel uncertainties in these analyses are based on the characteristics of installed instrumentation, the environmental conditions present at the ~~instrumentation's~~instrumentation's installed locations, and process conditions. A properly established setpoint initiates a plant protective action before the process parameter exceeds its analytical limit. This, in turn, ~~assures~~ensures that the transient will be avoided and/or terminated before the process parameters exceed the established safety limits.

Similar calculations and reviews are performed as necessary to verify the setpoints for functions that are not related to a safety limit or for ~~non-safety~~nonsafety systems or procedural action points for safety and ~~non-safety~~nonsafety systems.

2. Information to Be Reviewed

The information to be reviewed consists of: (1) a description of the setpoint program, procedures, and analytical results, (2) engineering information for the installed instrumentation, (3) supporting analyses, and (4) provisions and operating history, if available, for the instrument maintenance and calibration program.

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3. Acceptance Criteria

Setpoint Documentation

The following information on the ~~licensee/applicant's~~licensee's or applicant's setpoint program should be provided for review:

- Facility setpoint list identifying safety setpoints and ~~non-safety~~nonsafety setpoints for functions providing protective functions important to safety or that are relevant to compliance with technical specification limiting conditions for operation.

- Identification of safety setpoints that are not safety-limit-related LSSS and the basis for this determination.

- Identification of setpoints that trigger procedural actions that are important to safety.

- Description of the setpoint methodology and procedures used in determining setpoints, including information sources, scope, assumptions, interface reviews, and statistical methods.

- Terminology used to describe limits, allowances, and tolerances, and environmental or other effects used to support setpoint calculations.

- Technical specifications and the basis for LSSSs.

- Basis for acceptable as-found band and acceptable as-left band and determination of the instrument operability based on the acceptable as-found band and acceptable as-left band.

- Basis for calibration intervals.

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- Basis for assumptions regarding instrument uncertainties and a discussion of the method used to determine uncertainty values.

- Description of the provisions for the control of measuring and test equipment used for calibration of the instrument.

- Description of the program and methodology used to monitor and manage instrument uncertainties, including drift.

A documented basis for the safety~~-~~-system setpoint should be available for ~~Staff~~staff review.

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Documentation should conform ~~with~~to the guidance ~~of Regulatory Guide~~in RG 1.105~~, Revision 3.~~.

The description of the instrument channel in accordance with ISA-S67.04-1994, Part I, should include:

- Description of the functional and performance criteria for the initiation and execution of the safety functions at the setpoints.
- Instrument specifications, including range, accuracy, repeatability, hysteresis, dynamic response, environmental qualification, calibration reference, and calibration intervals for each instrument type.
- Instrument loop diagrams showing all hardware elements of the instrument loop(s).
- Instrument and tubing layout drawings and installation details showing locations and elevations of instruments and tubing relative to a reference datum, as well as the points where the instrument interfaces with the monitored process.
- For digital instrumentation, the configuration database for the instrumentation functions, and identification of digital elements (hardware and software) where error could be introduced into the measurement-for example, errors that could result from analog-to-digital or digital-to-analog conversion or from numerical methods used in the software (e.g., curve fitting).

The description of assumptions in accordance with ISA-S67.04-1994, Part I, should include the environmental allowances (temperature, pressure, humidity, radiation, vibration, seismic, and electrical) for the instruments.

Analysis Supporting Establishment of Setpoints and Instrumentation Tolerances

The applicant~~/~~ or licensee should document the bases and the calculations of measurement uncertainties. The methods by which setpoints are calculated should conform to the guidance ~~of Regulatory Guide~~in RG 1.105~~, Revision 3~~.

Statistical Guidelines for Instrument Uncertainty

In the review of uncertainties in determining a trip setpoint and its allowable values, the NRC staff typically uses 95/95 tolerance limits as an acceptable criterion, i.e., a 95 percent probability that the constructed limits contain 95 percent of the population of interest for the surveillance interval selected.

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Guidelines for Graded Approach

ISA-S67.04~~-1994~~, Part I, Section 4 states that the safety significance of various types of setpoints important to safety may differ, and thus a less rigorous setpoint determination method for certain functional units and limiting conditions of operation may be applied. The use of a graded approach allows a less rigorous setpoint determination method based on the safety significance of the instrument function. However, the grading technique chosen by the applicant~~/~~ or licensee should be consistent with the standard and should consider and bound all known applicable uncertainties regardless of setpoint application. Additionally, the application of the standard using a graded approach is also appropriate for ~~non-safety~~nonsafety system instrumentation maintaining design limits in the technical specifications.

Basis for Instrument Calibration Intervals

The applicant~~/~~ or licensee should evaluate the effects of extended calibration intervals on instrument uncertainties, equipment qualification, and vendor maintenance provisions to assure that an extended surveillance interval does not result in exceeding the assumptions stated in the safety analysis. Generic Letter 91-04, Enclosure 2, ~~"Guidance for Addressing the Effect of Increased Surveillance Intervals on Instrument Drift and Safety Analysis Assumptions,"~~ provides acceptable guidance for justifying extended calibration intervals through the use of data analysis, monitoring, and assessment. This approach has been used for plants to accommodate a 24-month fuel cycle change. For changes to surveillance test intervals for reasons other than a 24-month fuel cycle, the submittals have followed the risk informed approach and followed the guidance of ~~Regulatory Guides~~RGs 1.174, 1.177, and 1.200.

4. Review Procedures

The setpoint analysis methodology and assumptions should be reviewed to confirm that an acceptable analysis method is used and that the analysis parameters and assumptions are consistent with the safety analysis, system design basis, technical specifications, plant design, and expected maintenance practices. The following factors should be emphasized in the review:

- Relationships between the safety limit, the analytical limit, the limiting trip setpoint, the allowable value, the setpoint, the acceptable as-found band, the acceptable ~~asleft~~as- left band, and the setting tolerance.

- The reviewer should assure that the setpoint technical specifications meet the requirements of 10 CFR 50.36. Additional information related to setpoint technical specifications is provided in RIS 2006-17.

- Basis for selection of the trip setpoint.

- Uncertainty terms that are addressed.

- Method used to combine uncertainty terms.

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- Justification of statistical combination.
- Relationship between instrument and process measurement units.
- Data used to select the trip setpoint, including the source of the data.
- Assumptions used to select the trip setpoint (e.g., ambient temperature limits for equipment calibration and operation, potential for harsh accident environment).
- Instrument installation details and bias values that could affect the setpoint.
- Correction factors used to determine the setpoint (e.g., pressure compensation to account for elevation difference between the trip measurement point and the sensor physical location).
- Instrument test, calibration or vendor data, as-found and as-left; with each instrument should be demonstrated to have random drift by empirical and field data~~. Evaluation~~ and evaluation results should be reflected appropriately in the uncertainty terms, including the setpoint methodology.

The design, installation, calibration procedures, and calibration activities for specific channels may be inspected to gain further confidence that setpoint calculations are consistent with plant equipment and calibration procedures. NRC Inspection Manual, Procedure 93807, ~~"~~"Systems Based Instrumentation and Control Inspection~~,"~~," provides guidance for such inspections.

C. REFERENCES

1. ANSI/NCSL ~~Std.~~ Z540-1-1994, ~~"~~"Calibration Laboratories and Measuring and Test Equipment ~~-~~ - General Requirements~~."~~."

2. GL 91-04, "Changes in Technical Specification Surveillance Intervals to Accommodate a 24-Month Fuel Cycle."

~~2.~~3. IEEE Std~~.~~ 279-1971, ~~"~~"Criteria for Protection Systems for Nuclear Power Generating Stations~~."~~."

~~3.~~4. IEEE Std~~.~~ 498-1990, ~~"~~"IEEE Standard Requirements for the Calibration and Control of Measuring and Test Equipment Used in Nuclear Facilities~~."~~."

~~4.~~5. IEEE Std~~.~~ 603-1991, ~~"~~"IEEE Standard Criteria for Safety Systems for Nuclear Power Generating Stations~~."~~."

~~5.~~6. IEEE Std~~.~~ 7-4.3.2~~-2003, "~~, "IEEE Standard Criteria for Digital Computers in Safety Systems of Nuclear Power Generating Stations~~."~~."

~~6.~~7. ISA-S67.04~~-1994~~, Part I, ~~"~~"Setpoints for Nuclear Safety-Related Instrumentation Used in Nuclear Power Plants~~."~~."

~~7.~~8. ISA-S67.04-1994, Part II, “Methodology for the Determination of Setpoints for Nuclear Safety-Related Instrumentation~~.”~~.”

~~8.~~9. NRC Inspection Manual, Inspection Procedure 93807, ~~"~~“Systems Based Instrumentation and Control Inspection~~," U.S. Nuclear Regulatory Commission, May 31, 1994.~~.”

~~Regulatory Guide~~

~~9.~~10. RG 1.105, ~~Revision 3, "~~“Setpoints for Safety-Related Instrumentation~~," Office of Nuclear Regulatory Research, U.S. Nuclear Regulatory Commission, 1999.~~.”

~~Regulatory Guide~~

~~10.~~11. RG 1.152, ~~Revision 2,~~ “Criteria for ~~Digital~~Use of Computers in Safety Systems of Nuclear Power Plants~~,” Office of Nuclear Regulatory Research, U.S. Nuclear Regulatory Commission, January 2006.~~.”

~~Regulatory Guide~~

~~11.~~ RG 1.174, ~~Revision 1,~~ “An Approach for Using Probabilistic Risk

12. Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis~~,” Office of Nuclear Regulatory Research, U.S. Nuclear Regulatory Commission, November 2002.~~.”

~~Regulatory Guide~~

~~12.~~13. RG 1.177, “An Approach for Plant-Specific, Risk-Informed Decision Making: Technical Specifications~~,” Office of Nuclear Regulatory Research, U.S. Nuclear Regulatory Commission, August 1998.~~.”

~~Regulatory Guide~~

~~13.~~14. RG 1.200 (For Trial Use), “An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities~~,” Office of Nuclear Regulatory Research, U.S. Nuclear Regulatory Commission, February 2004.~~.”

~~14. Generic Letter 91-04, “Guidance on Preparation of a Licensee Amendment Request for Changes in Surveillance Intervals to Accommodate a 24-Month Fuel Cycle,” Office of Nuclear Reactor Regulation, U.S. Nuclear Regulatory Commission, April 2, 1991.~~

~~15.~~ NRC Regulatory Issue Summary 2006-17, “NRC Staff Position on the Requirements of

15. 10 CFR 50.36, ~~“~~“Technical Specifications,~~”~~” Regarding Limiting Safety System Settings During Periodic Testing and Calibration of Instrument Channels~~,” Office of Nuclear Reactor Regulation, U.S. Nuclear Regulatory Commission, August 24, 2006.~~.”

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PAPERWORK REDUCTION ACT STATEMENT

The information collections contained in the Standard Review Plan are covered by the requirements of 10 CFR Part 50 ~~and~~, and were approved by the Office of Management and Budget, approval number 3150-0011.
~~10 CFR Part 52, and were approved by the Office of Management and Budget, approval number 3150-0011 and 3150-0151.~~

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BTP Section 7-12
Description of Changes

BTP 7-12, "Guidance on Establishing and Maintaining Instrument Setpoints"

This BTP Section affirms the technical accuracy and adequacy of the guidance previously provided in BTP 7-12, Revision 5, dated March 2007. See ADAMS Accession Number ML070550078.

The
main purpose of this update is to incorporate the revised software Regulatory Guides and the associated endorsed standards. For organizational purposes, the revision number of each Regulatory Guide and year of each endorsed standard is now listed in one place, Table 7-1. As a result, revisions of Regulatory Guides and years of endorsed standards were removed from this section, if applicable. For standards that are incorporated by reference into regulation (IEEE Std 279-1971 and IEEE Std 603-1991) and standards that have not been endorsed by the agency, the associated revision number or year is still listed in the discussion. Additional changes were editorial.

Part of 10 CFR was reorganized due to a rulemaking in the fall of 2014. Quality requirement discussions in the former 10 CFR 50.55a(a)(1) were moved to 10 CFR 50.54(jj) and 10 CFR 50.55(i). The incorporation by reference language in the former 10 CFR 50.55a(h)(1) was moved to 10 CFR 50.55a(a)(2). There were no changes either to 10 CFR 50.55a(h)(2) or 10 CFR 50.55a(h)(3).

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