U.S. NUCLEAR REGULATORY COMMISSION

STANDARD REVIEW PLAN

BRANCH TECHNICAL POSITION 7-10

GUIDANCE ON APPLICATION OF REGULATORY GUIDE 1.97

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, “Regulatory Requirements, Acceptance Criteria, and Guidelines for Instrumentation and Control Systems Important to Safety,” (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.

Revision 6 - August 2016

A. BACKGROUND

This branch technical position (BTP) provides additional guidelines for reviewing an applicant's or licensee's accident monitoring instrumentation. These guidelines are based on reviews of design submittals by applicants or licensees that contained approved interpretations and alternatives for the guidance identified in RG 1.97.

Revisions 2, 3, and 4 of RG 1.97 are currently in effect. Plants licensed before June 2006 committed to follow either Revision 2 or 3, both titled "Instrumentation for Light Water-Cooled Nuclear Power Plants to Assess Plant and Environs Conditions During and Following an Accident." Both Revisions 2 and 3 of RG 1.97 prescribe a detailed list of variables to monitor and specify a comprehensive list of design and qualification criteria to be met by the instrumentation monitoring each variable. Revision 4, "Criteria for Accident Monitoring Instrumentation for Nuclear Power Plants," endorses, with exceptions and clarifications, Institute of Electrical and Electronics Engineers (IEEE) Standard (Std) 497, "IEEE Standard Criteria for Accident Monitoring Instrumentation for Nuclear Power Generating Stations." IEEE Std 497 establishes flexible, performance-based criteria for the selection, performance, design, qualification, display, and quality assurance of accident monitoring variables.

1. <u>Regulatory Basis</u>

Title 10 of the *Code of Federal Regulations* (10 CFR) 50.34(f)(2)(xvii) requires in part that instrumentation be provided to measure, record, and read out in the control room: containment pressure, containment water level, containment hydrogen concentration, containment radiation intensity (high-level), and noble gas effluents.

10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities," 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 over their anticipated ranges for accident conditions, as appropriate, to ensure adequate safety.

GDC 19, "Control Room," requires in part that a control room be provided from which actions can be taken to maintain the nuclear power unit in a safe condition under accident conditions, including loss-of-coolant accidents. It also requires that equipment, including the necessary instrumentation, be provided at appropriate locations outside the control room and that such equipment have a design capability for prompt, hot shutdown of the reactor.

GDC 64, "Monitoring Radioactivity Releases," requires in part that means be provided to monitor (1) the reactor containment atmosphere, (2) spaces containing components for recirculation of loss-of-coolant accident fluids, (3) effluent discharge paths, and (4) the plant environs for radioactivity that may be released from postulated accidents.

2. <u>Relevant Guidance</u>

RG 1.97 describes a method acceptable to the U.S. Nuclear Regulatory Commission (NRC) staff for providing instrumentation to monitor variables for accident conditions.

Plants that obtained an operating license after June 2006 should reference the guidance of RG 1.97, Revision 4.

For plants with operating licenses issued before June 2006, RG 1.97, Revisions 2 and 3, are still effective. Licensees of these plants may, however, convert to the criteria of Revision 4 or use the criteria of Revision 4 when making modifications that do not involve a conversion. The guidance contained in Regulatory Position 1 of RG 1.97, Revision 4, should be followed in these cases. See the “Conversion to Revision 4” discussion in section B.3 of this BTP for additional information.

3. Purpose

The purpose of this BTP is to provide additional guidance for NRC reviewers to verify that the previously cited regulatory bases are met by an applicant’s or licensee’s submittal. This BTP’s objectives are:

- Clarify the staff position on accident monitoring instrumentation, and

- Identify alternatives acceptable to the staff for satisfying the guidelines identified in RG 1.97.

B. BRANCH TECHNICAL POSITION

1. Introduction

Applicants or licensees have provided design submittals to the staff containing interpretations of guidelines identified in RG 1.97. In some cases, applicants or licensees have requested relief from selected guidelines. When the applicants or licensee provided adequate justification, the staff has accepted alternatives to implementing specific provisions of Revisions 2 and 3 of RG 1.97. The NRC staff documented the basis for this acceptance in various safety evaluation reports (SERs). Staff positions and clarifications applicable to various classes of plant designs emerged from these evaluations. These positions include identification of specific designs acceptable to the staff for instrumentation to assess plant and environs conditions during and following an accident.

2. Information to Be Reviewed

Standard Review Plan (SRP) Section 7.5, “Information Systems Important to Safety,” describes the information to be reviewed for accident monitoring instrumentation.

3. Acceptance Criteria

The design and qualification criteria identified in RG 1.97 should be supplemented by the considerations outlined below:

Environmental Qualification

The staff has interpreted 10 CFR 50.49(b)(3), “Certain Post-Accident Monitoring Equipment,” as follows:

> For plants using Revisions 2 or 3 of RG 1.97, accident monitoring equipment that falls within the scope of Category 1 or 2 equipment should be environmentally qualified as

required by 10 CFR 50.49, “Environmental Qualification of Electric Equipment Important to Safety for Nuclear Power Plants,” or the applicant or licensee should provide an acceptable alternative for complying with 10 CFR 50.49(b)(3).

For plants using Revision 4 of RG 1.97, accident monitoring equipment identified as Type A, B, or C in accordance with that guide should be environmentally qualified as required by 10 CFR 50.49. Type D variables should be environmentally qualified for the particular accident‘s postulated environment at the installed location in accordance with the plant’s licensing basis. Licensees converting to Revision 4 or performing modifications based on Revision 4 may reference previously accepted alternatives as their basis for deviations from the environmental qualification criteria in Revision 4.

Seismic Qualification

For plants using Revisions 2, 3, or 4 of RG 1.97, if a reactor’s licensing basis does not include a commitment to RG 1.100, “Seismic Qualification of Electric Equipment for Nuclear Power Plants,” and credit is taken for original equipment in meeting the guidelines identified in RG 1.97, installation of the original equipment in conformance with the licensing basis for seismic qualification is acceptable, provided the other guidelines identified in RG 1.97 and this BTP are satisfied. However, for all reactors, new instrumentation that is installed to satisfy RG 1.97 or to replace original equipment for which credit was taken in satisfying RG 1.97 should satisfy the seismic qualification guidelines identified in RG 1.100.

Redundancy

For Category 1 variables under Revisions 2 and 3 of RG 1.97, no single failure should prevent the operators from being presented information necessary to determine the safety status of the plant and to maintain the plant in a safe condition following an accident. Channels provided to monitor a Revision 2 or 3 Category 1 variable do not need to meet this criterion during channel maintenance, test, or calibration, provided the duration of such testing satisfies the applicable requirements of the licensing basis. For example, the time interval required for a test, calibration, or maintenance operation could be shown to be so short that it would have an insignificant effect on the overall availability of the accident monitoring instrumentation system.

Independence of Redundant Instrumentation

For plants using Revisions 2, 3, or 4 of RG 1.97, if a reactor licensing basis does not include a commitment to RG 1.75, “Criteria for Independence of Electric Systems,” and credit is taken for original equipment in meeting the guidelines identified in RG 1.97, installation of the original equipment in conformance with the licensing basis requirements for separation and independence is acceptable, provided the other guidelines identified in RG 1.97 and this BTP are satisfied. However, for all reactors, new instrumentation that is installed to satisfy RG 1.97 or to replace original equipment for which credit was taken in satisfying RG 1.97 should satisfy the separation and isolation guidelines in RG 1.75.

Display and Recording

Revisions 2 and 3 of RG 1.97 state in part that if direct or immediate trend or transient information is essential for operator information or action, the recording should be continuously available on dedicated recorders. Otherwise, the information may be continuously updated, stored in computer memory, and displayed on demand. For the latter non-essential applications, the use of Category 2 computers or dedicated Category 2 recorders is acceptable for recording Category 1 information, provided the Category 1 instrumentation is isolated from the Category 2 instrumentation using qualified isolation devices. Guidance on the application and qualification of isolation devices is provided in SRP BTP 7-11, "Guidance on Application and Qualification of Isolation Devices."

Range

Deviations from the range values identified in Revisions 2 and 3 of RG 1.97 may be acceptable if supported by analyses demonstrating that the indication would remain on scale with appropriate margins for any design basis event or accident for which the instrumentation might be required for operator information. An appropriate margin should include allowance for analytical uncertainties and instrumentation uncertainties. However, RG 1.97 identifies that, for a limited number of functionally significant variables (e.g., containment pressure, primary system pressure); instrument ranges should extend beyond values that the selected variables can attain under limiting conditions. Guidance on uncertainties is provided in SRP BTP 7-12, "Guidance on Establishing and Maintaining Instrument Setpoints."

The ranges and footnotes for radiation and meteorological instrumentation that are provided in Revision 3 of RG 1.97 should be applicable for plants using Revision 4 of RG 1.97. Applicants and licensees using Revision 4 should document differences from the Revision 3 ranges and footnotes for radiation and meteorological instrumentation.

Minimizing Measurements

To the extent practicable, the same instruments should be used for accident monitoring as are used for normal operations of the plant. In cases in which a single display may indicate the reading of more than one instrument, the intent of this recommendation is met if the same variable and same display are used for accident monitoring even though the sensor(s) providing the signal are different.

Alternate Instrumentation

The use of alternate instrumentation to monitor variables different from those identified in RG 1.97, Revisions 2 and 3, is acceptable, provided that all three of the following criteria are met:

(1) The alternate instrumentation fulfills the purpose of the variables identified in RG 1.97.

(2) The alternate instrumentation conforms to the design and qualification criteria for the variables identified in RG 1.97.

(3) No credit is taken by the applicant or licensee in post-accident procedures, emergency operating procedures, or functional recovery guidelines for indication of the variables identified in RG 1.97 for which the alternative instrumentation is proposed.

Revision 4 of RG 1.97 does not identify specific variables to be displayed; therefore, this topic does not apply to Revision 4.

Guidance for Boiling Water Reactor (BWR) and Pressurized Water Reactor (PWR) Variables

Table 1, BWR Variables, and Table 2, PWR Variables, of RG 1.97, Revision 2, and Table 2, BWR Variables, and Table 3, PWR Variables of RG 1.97, Revision 3, identify guidelines for the range, design and qualification category, and purpose for specific BWR and PWR variables. For selected BWR and PWR variables identified in the tables, acceptable deviations or clarifications are identified in Tables 1 and 2, respectively, of this BTP. Tables 1 and 2 of this BTP list RG 1.97 variables and types of deviations from RG 1.97 guidelines (e.g., deviations with respect to category, redundancy, range, direct measurement), and provide a summary of the acceptance guidelines or clarification associated with the deviations. Revision 4 of RG 1.97 does not identify specific variables; therefore Tables 1 and 2 of this BTP do not apply to Revision 4.

Conversion to Revision 4

Applicants or licensees of reactors that are committed to either Revision 2 or 3 of RG 1.97 may voluntarily convert the plant's entire accident monitoring program to the criteria of Revision 4 of RG 1.97. Conversion means revising the commitment for the plant's entire accident monitoring program from the current licensing basis (either Revision 2 or 3) to the guidance in Revision 4. The conversion to Revision 4 could include physical changes (e.g., replacing an instrument), licensing changes (e.g., technical specification changes), changes in variable types and associated design and qualification criteria, changes in the function or purpose of the variable, and changes in the range being monitored. This conversion should be supported by a complete analysis of the plant's accident monitoring program against all of the criteria in Revision 4.

The applicant or licensee should document the results of the analysis in a table format that includes for each variable the following: variable name, current type, current function or purpose, current range, proposed type, proposed function or purpose, proposed range, and any criteria that would be changed. For variables for which there is a proposed change in type or purpose, the applicant or licensee should document the rationale for the change.

Modifications Using Revision 4

Applicants or licensees of reactors that are committed to either Revision 2 or 3 of RG 1.97 may voluntarily use the criteria of Revision 4 of RG 1.97 to perform modifications that do not involve a conversion. The applicant or licensee should first perform an analysis to determine the complete list of variables and their associated types in accordance with the selection criteria of Revision 4. Without such an analysis, there would not be a means to correlate Revision 4 criteria being applied to the modification of variables that have been previously licensed to the criteria of Revision 2 or 3.

The applicant or licensee should document the proposed modifications in a table format that includes for each variable the following: variable name, current type, current function or purpose, current range, proposed type, proposed function or purpose, proposed range, and any criteria that would be changed. For variables for which there is a proposed change in type or purpose, the applicant or licensee should document the rationale for the change.

Licensees may make modifications within the plant's current licensing basis without referencing Revision 4 of RG 1.97.

4. Review Procedures

SRP Section 7.5, “Information Systems Important to Safety,” Section III, describes the review procedure for accident monitoring instrumentation.

Table 1. For BWRs: Acceptable Deviations and Clarifications to Revisions 2 and 3 of Regulatory Guide 1.97

Variable	Deviation	Acceptance Guidelines/Clarification
Neutron flux	• Category • Equipment qualification • Redundancy • Power source • Quality assurance • Range	Except for applications submitted after January 13, 1993 (which should satisfy the guidelines identified in RG 1.97), the design criteria identified in NEDO-31558-A, "Position on Nuclear Regulatory Commission (NRC) RG 1.97, Revision 3, Requirements for Post-Accident Neutron Monitoring System," are an acceptable alternative to Category 1 criteria. Pursuant to these alternative criteria, the applicant/licensee should perform a plant-specific evaluation of the electrical power distribution to the neutron monitoring system (including the recorders) to verify that the instrument power is not lost during design basis events.
Coolant level in reactor vessel	• Range • Redundancy	If redundant channels of Category 1 instrumentation cover the fuel zone and the wide range (i.e., all manual and automatic trip functions), a single channel of Category 3 upset range instrumentation (from the upper end of the wide range to the top of the vessel or centerline of the main steam line) is acceptable for detection of water carryover.
Core temperature	N/A	RG 1.97 Rev. 2 & 3 Table 2, "BWR Variables," lists core temperature as a Type B and Type C variable used to provide diverse indication of reactor vessel water level. However, NUREG-0737, Supplement 1, "Clarification of TMI Action Plan Requirements: Requirements for Emergency Response Capability," Section 6.1b later clarified, "BWR incore thermocouples and continuous offsite dose monitors are not required pending their further development and consideration as requirements." The aforementioned development and considerations were not complete as of the publication of Rev. 3 of RG 1.97. Therefore, it is an acceptable deviation to not include Core Temperature as measured by qualified incore thermocouples as a RG 1.97 Rev. 2 or Rev. 3 variable.
Drywell sump and drywell drain sumps level	• Category • Direct measurement	Category 3 instrumentation (e.g., flow instrumentation) is an acceptable alternative to Category 1 instrumentation for this variable if it can be shown that: (a) For small leaks, the alternate instrumentation will not experience a harsh environment; (b) For larger leaks, the sumps fill promptly and the sump drain lines isolate due to the increase in drywell pressure, negating the need for the measurement; (c) Drywell pressure and temperature indication can be used to detect leakage into the drywell; and (d) The instrumentation neither automatically initiates nor alerts the operator to initiate operation of a safety system in a post-accident situation.
Primary containment isolation valve position	Redundancy	A redundant position indication for each active containment isolation valve is not necessary because the valves are redundant. Likewise, position indication is not necessary for valves within the operator's cognizance that are normally closed and remain closed after an accident, and that are administratively controlled.
Primary system safety relief valve positions, including automatic depressurization system (ADS) or flow through pressure in valve lines	Category	Category 3 safety relief valve (SRV) position indication is an acceptable alternative to Category 2 instrumentation for this variable, provided that a plant-specific evaluation documents that the analysis of NEDO-33160, "Regulatory Relaxation for the Post Accident SRV [Safety Relief Valve] Position Indication System," applies to the plant and that the requirements of NUREG-0737, Item II.D.3, and 10 CFR 50.34(f) continue to be met.

Table 1. For BWRs: Acceptable Deviations and Clarifications to Revisions 2 and 3 of Regulatory Guide 1.97

Variable	Deviation	Acceptance Guidelines/Clarification
Radioactivity concentration or radiation level in circulating primary coolant	N/A	A continuous post-accident monitor is not necessary.
Containment and drywell hydrogen concentration	Category	In accordance with 10 CFR 50.44, Category 3 instrumentation is acceptable for diagnosing the course of beyond design basis accidents.
	Range	In plants for which credit is taken for Class 1E hydrogen ignitors, the range recommendations may be relaxed if analysis shows that the instrumentation will remain on scale through all design basis events with adequate margin for uncertainties.
Containment and drywell oxygen concentration (inerted containment)	Category	In accordance with 10 CFR 50.44, Category 2 instrumentation is acceptable for verification of the status of an inerted containment.
	Range	The range recommendation may be relaxed if it can be shown that the instrumentation will perform adequately during all accident and post-accident conditions.
Suppression chamber and drywell spray flows	Direct measurement	The use of residual heat removal (RHR) flow, suppression chamber temperature and pressure, and drywell temperature and pressure are acceptable alternatives if it can be shown that (1) use of these variables can accurately and reliably measure the effectiveness of the drywell and suppression chamber spray in a timely manner, and (2) the position of the spray throttling valves can be monitored and the sprays adequately controlled from the control room using the alternative variables.
Standby liquid control system (SLCS) flow	Direct measurement	Measurement of SLCS pump discharge pressure and SLCS storage tank level may be acceptable as an alternative indication that the SLCS pump is operating and that SLCS flow is occurring.
Reactor building or secondary containment area radiation	Category	Area radiation monitors located in Mark III containments and in primary containments of other BWRs may be Category 2 as an alternative to Category 1 monitors. Area radiation monitors located in reactor building secondary containments for Mark I and Mark II plants and in other plant areas may be Category 3 in lieu of Category 2.
Radiation exposure rate/variables used to monitor airborne radioactive materials released from plant	Category	If the instrument is located in a mild environment and is not part of a safety system, Category 3 instrumentation is acceptable in place of Category 2 instrumentation.

Table 2. For PWRs: Acceptable Deviations and Clarifications to Revisions 2 and 3 of Regulatory Guide 1.97

Variable	Deviation	Acceptance Guidelines/Clarification
Neutron flux	Environmental qualification	A non-environmentally qualified instrument is acceptable if qualified core exit thermocouples and reactor coolant system (RCS) hot and cold leg temperature indications are provided in conjunction with directions in emergency procedures for operator action to ensure that boric acid injection is occurring.
RCS pressure (Combustion Engineering [CE] reactors)	Range	A range of 0-3,000 psig is an adequate alternative to 0-4,000 psig if analysis is presented or referenced in the final safety analysis report (FSAR) that shows that pressure will remain on scale for all design basis transients and accidents. However, if an anticipated transient without scram (ATWS) analysis indicates that pressures exceeding FSAR values are possible, an expanded range of the Category 1 instrumentation should be provided.
Containment sump level	Range	Separate narrow-range instrumentation is not required if the wide-range instrumentation satisfies the guidelines of RG 1.97 and is of sufficient range and accuracy to monitor the sump operation for all design basis conditions.
Containment isolation valve position	Redundancy	A redundant position indication for each active containment isolation valve is not necessary because the valves are redundant. Likewise, position indication is not necessary for valves within the operator's cognizance that are normally closed and remain closed after an accident, and that are administratively controlled.
Radioactivity concentration or radiation level in circulating primary coolant	N/A	A continuous post-accident monitor is not necessary.
Containment hydrogen concentration	Category	In accordance with 10 CFR 50.44, Category 3 instrumentation is acceptable for diagnosing the course of beyond design basis accidents.
	Range	In plants for which credit is taken for Class 1E hydrogen ignitors, the range recommendations may be relaxed if analysis shows the instrumentation will remain on scale through all design basis events with adequate margin for uncertainties.
Accumulator tank level and pressure	Category	The safety function of the accumulator is performed passively by opening the discharge check valve when RCS pressure is lower than the tank pressure. Therefore, Category 3 instrumentation is an acceptable alternative to Category 2 if there are no operator actions that depend on use of this instrumentation for accident mitigation.
Accumulator isolation valve position	Category	A Category 3 position indication is acceptable if the accumulator isolation valves are locked-open, and motor-operated valves (i.e., power to the motor operators is disabled during normal operation) cannot change position during an accident.
Pressurizer heater	Indication	At a minimum, status indication should be provided for pressurizer heaters governed by the technical

Table 2. For PWRs: Acceptable Deviations and Clarifications to Revisions 2 and 3 of Regulatory Guide 1.97

Variable	Deviation	Acceptance Guidelines/Clarification
status		specification (i.e., heaters required to be served by emergency power).
Quench tank temperature and pressure	Range	Pressure relief of the tank via rupture disk limits the temperature of the tank contents to saturated steam conditions (less than 750°F). Therefore, it is acceptable if the upper-range value includes (with adequate margin) the saturation temperature corresponding to the tank rupture disk relief pressure (e.g., a rupture disk relief pressure of 100 psig corresponds to 328°F saturation temperature). Likewise, an upper-range value less than the design pressure of the tank is acceptable if the upper-range value covers (with adequate margin) the rupture disk relief pressure.
Steam generator level (wide-range)	Redundancy	For the wide-range level, two-loop plants should have two channels of instrumentation per loop, but three- and four-loop plants may have one channel of instrumentation per loop.
Steam generator pressure	Redundancy	If steam generator pressure is identified as a Type A variable, two-loop plants should have two channels of instrumentation per loop, but three- and four-loop plants may have one channel of instrumentation per loop.
Containment atmosphere temperature	• Category • Direct measurement	Category 3 instrumentation is an acceptable alternative to Category 1 if it is shown that this instrumentation is considered to be backup instrumentation, (i.e., if containment atmosphere temperature is not used in any post-accident procedures, emergency procedures, or functional recovery guidelines, and Category 1 containment pressure instrumentation is available as primary instrumentation).
Containment sump water temperature	Direct measurement	As an alternative to Category 2 containment sump water temperature instrumentation, either Category 2 residual heat removal heat exchanger inlet or outlet temperature instrumentation is an acceptable alternative for determining containment cooling status. In plants for which the containment cooling function is provided by the recirculation spray system, either Category 2 recirculation spray system heat exchanger inlet or outlet temperature instrumentation is an acceptable alternative.
Makeup flow/letdown flow/ volume control tank (VCT) level	• Category • Direct measurement	Category 3 instrumentation is an acceptable alternative to Category 2 instrumentation if the charging and letdown lines are isolated with an accident signal and no credit is taken for indication of these variables in post-accident procedures, emergency procedures, or functional recovery guidelines.
Radiation exposure rate and variables used to monitor airborne radioactive materials released from plant	Category	If the instrument is located in a mild environment and is not part of a safety system, Category 3 instrumentation is an acceptable alternative to Category 2 instrumentation.

C. REFERENCES

1. General Electric Report NEDO-31558-A, "Position on NRC RG 1.97, Revision 3, Requirement for Post-Accident Neutron Monitoring System," March 1993.

2. General Electric Report, NEDO-33160-A, Revision 1, "Regulatory Relaxation for the Post Accident SRV Position Indication System," October 2006.

3. Institute of Electrical and Electronics Engineers, IEEE Std 497, "IEEE Standard Criteria for Accident Monitoring Instrumentation for Nuclear Power Generating Stations."

4. U.S. Nuclear Regulatory Commission, "Clarification of TMI Action Plan Requirements: Requirements for Emergency Response Capability," NUREG-0737, Supplement 1, January 1983.

5. U.S. Nuclear Regulatory Commission, "Seismic Qualification of Electric Equipment for Nuclear Power Plants," Regulatory Guide 1.100.

6. U.S. Nuclear Regulatory Commission, "Criteria for Independence of Electrical Safety Systems," Regulatory Guide 1.75.

7. U.S. Nuclear Regulatory Commission, "Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant and Environs Conditions During and Following an Accident," Regulatory Guide 1.97, Revision 2.

8. U.S. Nuclear Regulatory Commission, "Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant and Environs Conditions During and Following an Accident," Regulatory Guide 1.97, Revision 3.

9. U.S. Nuclear Regulatory Commission, "Criteria for Accident Monitoring Instrumentation for Nuclear Power Plants," RG 1.97, Revision 4.

10. U.S. Nuclear Regulatory Commission, Safety Evaluation by the Office of Nuclear Reactor Regulation, "Pressurized Water Reactors Accumulator Pressure and Volume Instrumentation - Relaxation of RG 1.97 Environmental Qualification Requirements," January 21, 1992.

11. U.S. Nuclear Regulatory Commission, Safety Evaluation by the Office of Nuclear Reactor Regulation, "Boiling Water Reactors, RG 1.97, Post-Accident Neutron Flux Monitoring Instrumentation," January 13, 1993.

12. U.S. Nuclear Regulatory Commission, Safety Evaluation by the Office of Nuclear Reactor Regulation, "Pressurized Water Reactors Containment Sump Water Temperature Instrumentation, RG 1.97," November 22, 1993.

13. U.S. Nuclear Regulatory Commission, Safety Evaluation by the Office of Nuclear Reactor Regulation, "Final Safety Evaluation for Boiling Water Reactor Owners' Group (BWROG) Topical Report (TR) NEDO-33160, Regulatory Relaxation for the Post Accident SRV [Safety Relief Valve] Position Indication System," September 25, 2006.

BTP Section 7-10
Description of Changes

BTP 7-10, “Guidance on Application of Regulatory Guide 1.97”

This BTP Section affirms the technical accuracy and adequacy of the guidance previously provided in BTP 7-10, Revision 5, dated March 2007. See ADAMS Accession No. ML070550082.

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.

Added additional background information to Table 1 for the “Core Temperature” variable and reformatted Tables 1 and 2.

Additional changes to this BTP section in this revision were editorial.