NRC Staff Responses to Public Comments on DG-1335,
"Criteria for Accident Monitoring Instrumentation for Nuclear Power Plants"
(82 FR 61043; December 26, 2017)

I. INTRODUCTION

This document presents the U.S. Nuclear Regulatory Commission's (NRC's) responses to written public comments received on Draft Regulatory Guide DG-1335, "Criteria for Accident Monitoring Instrumentation for Nuclear Power Plants," issued December 2017 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML17083A134), in response to the separate notice published in Volume 82 of the *Federal Register*, page 61043 (82 FR 61043; December 26, 2017).

II. OVERVIEW OF COMMENTERS AND COMMENTS

The staff received 8 comment submissions and a total of 43 individual comments. Table 1 presents information on the commenters who submitted comments on DG-1335.

Table 1 Commenters Who Submitted Comments on DG-1335

Name	Affiliation	ADAMS Accession No.	Identifier
Patricia Campbell	GE Hitachi Nuclear Energy	ML18068A039	GEH
David Herrell	N/A	ML18068A040	DH
Alexander Klemptner	DTE Energy	ML18068A041	DTE
Gary Johnson	IEEE Member	ML18068A043	IEEE1
Anonymous	N/A	ML18068A044	AN
Ken Schrader	PWR Owners Group	ML18068A045	PWROG
Greg Hostetter	IEEE Member	ML18068A046	IEEE2
Gary Peters	Framatome, Inc.	ML18068A047	FR

The NRC staff binned the comments into the groups below to facilitate the agency's responses. The parentheses that follow each group name contains the number(s) of the regulatory positions in Section C of the version of DG-1335 published for public comment.

a. Comments on the Scope and Regulatory Guide Applicability
b. Comments on the Conversion of the Postaccident Monitoring System (C1 and C2)
c. Comments on Type A Variables Associated with Contingency Actions (C3)
d. Comments on Range (C4)
e. Comments on Accuracy (C5)
f. Comments on Number of Measurement Points (C6)
g. Comments on Common-Cause Failures (C7)
h. Comments on Human Factors Engineering (C8)
i. Editorial Comments

a. Comments on the Scope and Regulatory Guide Applicability

Comment [AN]: Good.

NRC Response: The NRC acknowledges the comment. The comment does not propose a change or correction to Regulatory Guide (RG) 1.97, "Criteria for Accident Monitoring Instrumentation for Nuclear Power Plants." The NRC did not make any change to the RG as a result of this comment.

Comment [PWROG-8]: The applicability statement includes, "those who have permanently ceased operations and have certified that fuel has been permanently removed from the reactor vessel." This conflicts with the NRC, "Power Reactor Transition from Operations to Decommissioning Lessons Learned Report," dated October 2016 (ADAMS No. ML16085A029) Appendix, which states, "Once a licensee's certifications under 10 CFR 50.82(a)(1) are docketed by the NRC, the licensee is no longer authorized to operate the reactor or emplace or retain fuel in the reactor vessel, and the SFP becomes the primary safety concern for site personnel. In the event of a challenge to the safety of the stored fuel, decision makers would not have to prioritize actions, because there would be no fuel located in the reactor vessel and, thus, no concern for providing cooling to that location. Consequently, the NRC staff concluded that these showings provide good cause for rescission of the order."

NRC Response: The NRC disagrees with this comment and believes that this comment can be interpreted two ways:

(1) Interpreting the comment literally, does the draft RG (DG-1335) really conflict with the referenced document?

(2) Interpreting the comment presumptively, would such licensees be subject to the guidance in RG 1.97, Revision 5, pertaining to the provisions of Type F parameters for monitoring fuel damage and the effects of fuel damage, which might include fuel damage resulting from fuel-related accidents that occur within the spent fuel pool?

With regard to the first interpretation, the NRC believes that there is no actual conflict between the RG and the referenced "lessons learned" document. RG 1.97 provides guidance for accident monitoring instrumentation for a licensee that has permanently removed fuel from the vessel and permanently ceased power operations and that voluntarily chooses to adopt the new guidance (a highly unlikely scenario, since most licensees would likely elect to retain their existing licensing basis commitments with respect to conformance with RG 1.97 prior to ceasing operations). In contrast, the "lessons learned" document provides guidance identifying possible licensing actions that could occur subsequent to the ceasing of power operations and placed all remaining fuel into the spent fuel pool. The "lessons learned" document simply states that the NRC has good cause to approve a possible rescission of one or more of the post-Fukushima orders, once a licensee has requested rescission through the submittal of a license amendment request.

With regard to the second interpretation, if a licensee first voluntarily adopts the use of the proposed RG-1.97, Revision 5, and then ceases operations and offloads all fuel into the spent fuel pool, there is essentially no risk of design-basis accidents associated with operations of the reactor. However, any commitments made with regard to operations of the spent fuel pool would need to be evaluated on a case-by-case basis, depending on the licensee's final configuration and prior commitments.

Comment [DH-1]: 1) For clarity, replace "It should be used …" with "IEEE Std. 7-4.3.2 as endorsed by RG 1.152 should be used …" for clarity, if that is the intent. 2) Replace "… the he most recent version of IEEE Std. 603 …" with "… the endorsed revision of IEEE Std. 603 …" for

correctness, since the most recent version of IEEE Std. 603 is not endorsed. 3) Further, the sentence also references "… when a computer is to be used as a component of a …" PAMS. If technology other than a computer (e.g., microcontroller, microprocessor, or programmable logic such as field programmable gate array or FPGA or application specific integrated circuit or a ASIC) is used, is the intent that this section does not apply? Else, the reference (i.e., "computer") should be to some form of non-restrictive "programmable digital technology."

NRC Response: The NRC agrees with this comment. The staff has removed the related language.

Comment [PWROG-1]: SAMG instrumentation is a voluntary initiative and as such there should be no regulatory involvement for the existing fleet of plants.

NRC Response: The NRC agrees with the comment and has revised the clause to clarify the staff's position.

RG 1.97, Revision 5, is mainly intended for use by applicants of new or advanced reactors and does not impose any new requirements for the existing fleet of plants. Unless a licensee elects to perform a conversion to RG 1.97, Revision 5, the guidance in this revision does not apply. Some combined license applicants reference a design control document that requires the implementation of severe accident management guidelines (SAMGs). The new revision offers these applicants an acceptable way to address the instrumentation that will be needed.

Although the SAMGs are a voluntary initiative for most licensees, the Nuclear Energy Institute informed the NRC in an October 26, 2015, letter of an industry initiative to perform timely updates of site-specific SAMGs. The NRC staff, in accordance with the Commission's direction (SRM-SECY-15-0065 (ADAMS Accession No. ML15239A767)), will update the reactor oversight process to explicitly provide periodic oversight of industry's implementation of the SAMGs. The products to be considered as part of the oversight activity include (1) the industry's proposal for the staff's oversight of SAMGs, (2) the Pressurized Water Reactor Owners Group (PWROG) and the Boiling Water Reactor Owners Group generic SAMG technical guidance documents, and (3) the site-specific SAMGs that licensees have committed to update for consistency with the generic guidance. Hence, a new RG 1.97 revision is necessary to provide guidance for new applicants or the existing licensees that want to add new instrumentation or convert some of their existing instrumentation to Type F variables.

In addition, the endorsement of Institute of Electrical and Electronics Engineers (IEEE) Standard (Std.) 497-2016, "IEEE Standard Criteria for Accident Monitoring Instrumentation for Nuclear Power Generating Stations," is appropriate because the standard is an international consensus standard published by IEEE and directly adopted by the International Electrotechnical Commission (IEC) as IEC 63147:2017, "Criteria for Accident Monitoring Instrumentation for Nuclear Power Generating Stations."

Comment [PWROG-2]: It is unclear what Licensing action(s) would trigger a need to apply the new standard.

NRC Response: The NRC agrees in part with the comment and has revised RG 1.97 as appropriate. This revision of RG 1.97 endorses the use of IEEE Std. 497-2016, which is intended for use by new reactor combined license applicants using design certifications and operating licensees committed to implement SAMGs. Existing reactor licensees choosing to voluntarily convert their existing accident monitoring program to implement RG 1.97, Revision 5, would do so with a license amendment under Title 10 of the *Code of Federal Regulations* (10 CFR) 50.90, "Application for Amendment of License, Construction Permit, or Early Site Permit."

Please see the response to [PWROG-1].

Comment [PWROG-3]: Implementation of the standard for SAMG instrumentation should be completely voluntary and applying it for one type of instrumentation should not imply/require it be applied to others.

NRC Response: The NRC agrees with the comment in part and has revised RG 1.97 as appropriate. The response to [PWROG-1] answers this question. With regard to licensees voluntarily adopting the guidance in RG 1.97, Revision 5, the NRC expects them to first review the entire accident monitoring program. For the implementation of only the Type F variables, a licensee should first evaluate the basis underlying its severe accident analysis to determine the variables to be selected, and the licensee should indicate in its license amendment request how it applied each aspect of RG 1.97, Revision 5, that is different from the corresponding aspect of the RG 1.97 revision that applies to its current licensing basis commitments for accident monitoring.

Comment [PWROG-4]: Qualification of SAMG instrumentation is inherently not feasible as there is no guidance on determining the conditions which the instrumentation could experience.

NRC Response: The NRC disagrees with the comment and, therefore, did not make any changes to RG 1.97. The requirements and guidance for equipment qualification (seismic and environmental qualification) of accident monitoring instruments should be consistent with the assigned function of that variable during and following a design-basis event, a severe accident, or a seismic event. Although the NRC did not endorse References 20 and 21 of RG 1.97, they provide helpful information for determining various conditions that the instrumentation could experience.

Comment [IEEE1-1]: Last full paragraph in background, "The plants cannot do so without an equivalent to Type F variables…" It should be noted that many plants already use existing plant instruments for severe accident management. See EPRI TR-102371, EPRI TR-103412, and section 3.3 of IAEA NP-T-3.16. The incorporation of Type F variables into accident monitoring systems is expected to be an improvement over the use of the methods discussed in the above reports. Those methods are outside of the scope of IEEE Std. 497-2106. Nevertheless, retention of information about monitoring severe accidents using the methods described by EPRI would provide a useful backup to Type F monitoring channels.

NRC Response: The NRC agrees in part with this comment and has revised the related sentence in RG 1.97, Section B, as follows:

> Therefore, this guidance may be useful for new applications or for existing plant licensees who choose to add new or convert some of their existing instrumentation to Type F variables.

The NRC disagrees that the agency should include Electric Power Research Institute (EPRI) TR-102371, "Instrument Performance under Severe Accident Conditions: Ways to Acquire Information from Instrumentation Affected by an Accident," dated December 1, 1993; EPRI TR-103412, "Assessment of Existing Plant Instrumentation for Severe Accident Management," dated December 1, 1993; and Section 3.3 of International Atomic Energy Agency (IAEA) NP-T-3.16, "Application of Accident Monitoring Systems for Nuclear Power Plant," issued 2015, in RG 1.97, Section B, or as supporting guidance based on any useful information that may be found within these documents. The staff has concluded that adding a discussion of these documents in RG 1.97 could be interpreted as partial or full endorsement of these documents, whereas, as the comment states, "those methods are outside of the scope of IEEE Std. 497-2016." Therefore, the staff does not intend to add these documents as discussion elements or references within this RG.

Comment [PWROG-9]: Page 12, Use by NRC Staff states, "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 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." The RG does not define directly relevant or essential consideration which could result in unduly forcing licensees to comply with this regulatory guide for generic submittals such as; Subsequent Licensee Renewal submittals, Surveillance Frequency Control Program submittals, Defueled TS submittals, etc.. Either these terms should be defined or examples given as to when this compliance requirement is appropriate to be invoked by the NRC staff.

NRC Response: The NRC disagrees with the comment. The paragraph in RG 1.97, Section A, that provides the purpose of RGs was inadvertently omitted and has been added into the guidance. This paragraph states, "The NRC issues RGs to describe to the public methods that the staff considers acceptable for use in implementing specific parts of the agency's regulations, to explain techniques that the staff uses in evaluating specific problems or postulated events, and to provide guidance to applicants. Regulatory guides are not substitutes for regulations and compliance with them is not required. Methods and solutions that differ from those set forth in RGs will be deemed acceptable if they provide a basis for the findings required for the issuance or continuance of a permit or license by the Commission." RG 1.97, Section D, referenced in the comment includes additional clarifications with regard to expectations that relate to the use of RGs and is not in conflict with the purpose of RGs. Unless an RG is incorporated by reference into a rule, licensees can demonstrate compliance with the rule by voluntarily adopting the RG or by using methods or solutions that differ from those described in the RG if the NRC staff finds that the proposed methods or solutions provide sufficient basis and information.

b. <u>Comments on the Conversion of the Postaccident Monitoring System (C1 and C2)</u>

Comment [IEEE1-2]: The position is not unreasonable for licensees converting from RG 1.97 rev. 1, 2, 3 or versions of IEEE Std. 497 prior to 2002. For users of IEEE Std. 497-2002 or RG 1.97 rev. 4 it should only be necessary to review the existing analyses for variable types A to E make what changes are needed to bring the systems into compliance with IEEE Std. 497-2016.

NRC Response: The NRC agrees in part with the comment and has revised the RG as appropriate. The new Type F variables are also to be determined by the licensee, in addition to reviewing the analysis for variable Types A through E, and are specific for each nuclear power plant. As addressed in Staff Regulatory Positions 1 and 2 of Section C, if an applicant (or

licensee) has already performed a comprehensive analysis of existing accident monitoring variables, then the applicant needs to review, address, and document its analysis of the variables only if it wants to convert the accident monitoring program. If an applicant adds new Type F variables to support the development of SAMGs, then it needs to evaluate the basis underlying its severe accident analysis to determine the variables to be selected, address them, and then document its analysis of the Type F severe accident variables.

Comment [IEEE1-3]: The statement "The licensee or applicant should first perform a comprehensive analysis of severe accidents to determine the variables to be selected" is overkill.

Presumably applicants will, and licensees will have already, performed, a comprehensive analysis of severe accidents to support the development of SAMG. More reasonable language would be something along the line of "The licensee and applicant should review plant Severe Accident Management Guidance and the supporting analysis to determine the variables to be selected." But even this seems unnecessary, as similar words are given in IEEE Std. 497-2016 clause 4.6 and in the corresponding row of Table 1.

NRC Response: The NRC agrees in part with the comment and has revised the RG as appropriate. Please see the response to [IEEE1-2].

c. Comments on Type A Variables Associated with Contingency Actions (C3)

Comment [DH-3]: It is not clear from the discussion provided why the NRC staff changed the industry consensus standard from not requiring Type A classification of variables associated with contingency actions to requiring Type A classification. The rationale for this change is not provided before the "Therefore," and thus there is no reason provided to reverse the industry consensus standard's conclusion, especially when the NRC participated in the development of the industry consensus standard.

NRC Response: The NRC disagrees with this comment but has revised the text to clarify the staff's position.

The scope of variables analyzed as Type A should include those variables that are associated with specific planned manually controlled actions for which no automatic control is provided. The staff clarified the second to last sentence of Clause 4.1, "Type A Variable," of IEEE Std. 497-2016, which states:

> Type A variables do not include those variables that are associated with contingency actions that may also be identified in written procedures.

The NRC staff understands this to mean that Type A variables include those variables associated with specific, preplanned, manually controlled actions for which no automatic control is provided that may also 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)." IEEE Std. 497-2016, Clause 4.1, uses this definition to prescreen out potential Type A variables. The staff agrees with the criteria in this clause, except during the initial identification process. Nuclear steam supply system vendors have not used the "contingency actions" term consistently in emergency procedures guidelines for current plant

designs; therefore, the staff recommends not using the “contingency actions” term in accordance with the modified criteria in the second to last sentence of Clause 4.1. Furthermore, the variable selection process encompassed a nonprescriptive, performance-based approach in RG 1.97, Revision 4, issued June 2006, and this RG 1.97, Revision 5. Therefore, the staff cannot endorse the complete exclusion of variables based on the “contingency actions” term, especially those associated with plant-specific operating procedures or guidelines. Instead, the scope of instruments that could potentially be selected for accident monitoring (based on the selection criteria) should initially be as encompassing as possible. Later, in the process of selecting the final list of variables to be monitored, licensees could screen out those instruments associated only with those contingency actions designed to address conditions beyond the plant’s licensing basis.

Comment [DTE-3]: Changing the Type A variables application scope by adding the contingency actions would contradict the current LDB and Tech Spec requirements with respect to the PAM instrumentation at the operating plants. Per Clause 4.1, manual actions are taken in response to Type A variables information are taken during design basis events considered in the FSAR, while contingency actions are taken to mitigate plant conditions occurring as a result of multiple equipment failures.

NRC Response: The NRC disagrees with this comment; however, it has revised the text to clarify the staff’s position. Please see the response to [DH-3].

Comment [IEEE1-4]: This clause should be deleted. The wording of this sentence in IEEE Std. 497-2016 is identical to the wording in IEEE Std. 497-2002. This wording was endorsed by rev. 4 of RG 1.97. Thus inserting clause 3 into a rev. 5 would create a conflict between the two versions. It is unreasonable for NRC to have two active documents that completely contradict each other.

Furthermore, implementation of the NRC’s proposal may have an effect on plant technical specifications as the tech spec bases for including all Type A variables does not seem to include the variables that the draft Reg. Guide propose as additions. See for example the notes at the bottom of Table 3.3.3-1 in NUREG-1431 Vol. 1, Rev. 3 and the corresponding bases in Volume 2. Therefore, it would seem that a revision of the standard tech specs would also be required. Otherwise, licensees adopting the new guide would be faced with the need to apply for changes to the post-accident monitoring instrumentation tech specs.

NRC Response: The NRC disagrees with this comment; however, it has revised the text to clarify the staff’s position. Please see the response to [DH-3].

The wording of this sentence in IEEE Std. 497-2016 is identical to the wording in IEEE Std. 497-2002; however, RG 1.97 did not endorse this wording as stated in Staff Regulatory Position 4 of RG 1.97, Revision 4.

d. <u>Comments on Range (C4)</u>

Comment [DH-4]: In the first modification, replace “… shall have extended ranges, sufficient range to cover, with appropriate margin, the predicted limits…” with “… shall have sufficient range with appropriate margin to cover the predicted limits …” for clarity.

NRC Response: The NRC agrees with this comment and has revised the RG as appropriate.

Comment [IEEE2-1]: The clarification provided in this regulatory position is unnecessary because the terms "sufficient range" and "appropriate margin" add nothing without defining what is "sufficient" and "appropriate".

NRC Response: The NRC disagrees with this comment and, therefore, did not make any changes to the RG.

Because RG 1.97, Revision 5, will mainly be used for new and advanced reactors with a range of instrumentation that is plant specific, the phrase “sufficient range with appropriate margin” is proper and flexible to cover the range of instrumentation for existing and new designs.

Comment [IEEE1-6]: In this statement it would be clearer to make reference to the variable types. It is suggested to change the statement to “Type A and B variables should cover, with appropriate margin…”

NRC Response: The NRC disagrees with the comment and, therefore, did not make any changes to the RG.

The use of sufficient ranges should apply to all variable types, which are already covered in the previous revisions of RG 1.97, not just Types A and B. The staff position conveyed in RG 1.97 Revision 5 is specific to Type F variable, which is newly added in IEEE Std. 497-2016.

Comment [DH-5]: 1) In the second modification, this should be a separate, stand-alone paragraph, as it is inappropriate to group instruments for EOPs with instruments for Type F (SAMG) variables in the existing paragraph. 2) Further, normal practice is to spell out the text and provide the acronym in parenthesis such as: “Emergency Operating Procedures (EOPs)” and not “EOPs (Emergency Operating Procedures)”.

NRC Response: The NRC agrees with this comment and has revised the RG as appropriate. The staff has separated out the position on the range of EOP-related and SAMG-related instrumentation into its own position.

Comment [DTE-4]: Current plants include allowances for measurement errors in the LSSS calculations consistent with their setpoint methodology. Proposed EOP instrument range determination method should be further evaluated for consistency with this methodology.

NRC Response: The NRC agrees with the comment; however, it did not make any changes to the RG.

RG 1.97, Revision 5, is mainly used by applicants for new and advanced reactors and does not impose any new requirements for the existing fleet of plants. Unless a licensee elects to perform a conversion to RG 1.97, Revision 5, or only add Type F variables, the guidance in this revision does not apply. The licensee and applicant are responsible for coordinating the range, accuracy, and margins.

e. <u>Comments on Accuracy (C5)</u>

Comment [DH-6]: 1) The paragraph starting “In addition, IEEE Std. 497-2016, Annex A…” should be a separate regulatory position to avoid conflating Clause 5.2 with Annex A. 2) Further, the wording of the paragraph leaves the impression that the intent of this paragraph is to endorse the informative (not normative) Annex A?

NRC Response: The NRC agrees with this comment and has revised the RG as appropriate. The staff deleted this paragraph and incorporated the following new statement in RG 1.97, Section B:

> The NRC does not endorse any annex in IEEE Std. 497-2016, since those annexes are informative only.

f. Comments on the Number of Measurement Points (C6)

Comment [IEEE1-7]: Delete RG C6 since Clause 5.6 in IEEE Std. 497-2016 takes care of this point.

NRC Response: The NRC disagrees with this comment and, therefore, did not make any changes to the RG. Clause 5.6 in IEEE Std. 497-2016 mentions calibration uncertainties, loop errors, drift, and errors imposed by environmental and seismic conditions. Those errors are different from the errors resulting from the number of measurement points in Regulatory Position 6 of RG 1.97, Section C.

Comment [IEEE2-2]: This clarification is unnecessary as implementation of the design criteria in Clause 6 will guide the user to the appropriate number of points necessary to characterize the variable.

NRC Response: The NRC disagrees with this comment and, therefore, did not make any changes to the RG.

Please see the response to [IEEE1-7].

g. Comments on Common-Cause Failures (C7)

Comment [GEH-1]: IEEE Std. 379-2014 (Section 5.5) asserts that CCF is out of scope for that standard. IEEE Std. 603-2009 (Section 5.16) simply points to IEEE Std. 7-4.3.2 for digital systems. Thus, the guidance appears to indicate that a licensee or applicant would conclude that it should use BTP 7-19 as the guidance and not as an alternative. The NRC should clarify the intent (but see the comment below). DG-1335 Section C.7 refers to BTP 7-19 as guidance for addressing CCF of Type/Category A, B, or C accident monitoring functions. However, the criteria in BTP 7-19 do not appear to be directly applicable to accident monitoring instrumentation. While there is discussion in BTP 7-19 (Rev. 7, August 2016) regarding monitoring instrumentation and manual actions (see Section 3.10), the acceptance criteria in Section 3.1 through 3.9 would appear to be "Not Applicable." Thus, the guidance is unclear how the criteria of BTP 7-19 should be applied to accident monitoring instrumentation regarding CCF. The NRC should consider if guidance in BTP 7-19 should be expanded.

NRC Response: The NRC agrees with this comment in part and has removed this staff position.

Comment [DH-7]: This regulatory position should be replaced with a simple statement that 1) software common cause failure of post-accident monitoring equipment is not required, since the NRC's governing policy, SRM/SECY 03-087 [SIC: should be SRM/SECY 93-087], only applies to the Reactor Trip System and the Engineered Safety Features Actuation System and 2) hardware common cause failure is addressed by the equipment qualification program. Thus,

since common cause failure does not apply to post-accident monitoring equipment, the NRC should not endorse this Clause in the IEEE standard.

NRC Response: The NRC agrees with this comment in part and has removed this staff position.

Comment [IEEE1-9]: The intent of this clause is unclear. Is the position indicating that endorsed IEEE standards are not acceptable? Perhaps this paragraph is simply trying to say that the versions of 379 and 7-4.3.2 to be used are the ones corresponding to their current Reg. Guides and the version of 603 to be used is as explained in 10 CFR 50.55a(h). But, this idea is also stated in the bulleted paragraph on the use of secondary references. Perhaps this first clause in the sentence should be deleted.

NRC Response: The NRC agrees with this comment in part and has removed this staff position.

Comment [FR-2]: This section discusses the applicability of IEEE-379-2016 and IEEE-603-2009, and directs the applicant to verify the applicability of these standards. The NRC has not formally endorsed these standards. This makes it difficult, or impossible for an applicant to determine what would be an acceptable interpretation of the requirements contained in those standards. Remove the reference to IEEE and other standards not endorsed by the NRC, or clearly state they are not endorsed.

NRC Response: The NRC agrees with this comment in part and has removed this staff position.

Comment [FR-3]: 1) This section refers to BTP 7-19 and Diversity and Defense-in-Depth guidance. It is not clear how, or if, this applies to Accident Monitoring Systems, and what the regulatory basis is.

2) The position described in this Draft Regulatory Guide is not consistent with the SRM on SECY-93-087, Item 18, 11.Q.

3) BTP 7-19 and the SRM are not an adequate regulatory basis for requiring diversity in accident monitoring instrumentation.

4) Another example is the cited NRC Order EA-12-051, which contains requirements for Spent Fuel Pool Level Instrumentation. This order does not require diversity or consideration of common cause failure. Revise the draft regulatory guide to be consistent with current regulations and regulatory guidance.

NRC Response: The NRC agrees with this comment in part and has removed this staff position.

Comment [FR-4]: The common cause failure position in this document is not consistent with on-going NRC work that is being pursued as part of the NRC I&C Integrated Action Plan (IAP), Modernization Plan MP1. Hold the issue of the Regulatory Guide until such time it can be harmonized with the results of the MP1 work.

NRC Response: The NRC agrees with this comment in part and has removed this staff position.

Comment [FR-1]: This section states the following: "The design of instrumentation should incorporate diversity and defense-in-depth as part of addressing common cause failures." Provide the regulatory basis (GDC, Regulation, Regulatory Guide, CFR reference) for this statement. Do not cite "guidance."

NRC Response: The NRC agrees with this comment in part and has removed this staff position.

Comment [IEEE1-10]: Delete from "alternately" to the end of this paragraph. The intent of BTP 7-19 as it is applied to displays and controls is to respond to point 4 of the NRC's Four Point Position, "A set of displays and controls located in the main control room shall be provided for … monitoring the parameters that support the safety functions. The displays and controls shall be independent and diverse from the safety computer system …." Such an analysis is important to the DinD&D strategy for the protection system but it does not respond to the intent of 6.2 of IEEE Std. 497-2016, as that clause is dealing with potential common cause failure within the accident monitoring system.

NRC Response: The NRC agrees with this comment and has removed this staff position.

h. <u>Comments on Human Factors Engineering (C8)</u>

Comment [DH-8]: C. Staff Regulatory Guidance C.8: This regulatory position paraphrases the requirements in other RGs which endorse appropriate, complete guidance for Human Factors Engineering. Since the IEEE standard provides only a few concerns to be addressed specific to this instrumentation by the licensee's or vendor's separate human factors engineering program, this entire regulatory position adds no value, has the potential to introduce confusion, and thus should be deleted. The statements provided in this regulatory position provide no additional information, and even referencing the RGs, NUREGs, and Standard Review Plan guidance for human factors serves no useful goal for a standard defining needs for post-accident monitoring and instrumentation.

NRC Response: The NRC agrees that this staff position can be deleted. However, Clause 5.3, "Response Time"; Clause 8.1.2, "Human Factor"; and Clause 8.5, "Display Location," of IEEE Std. 497-2016 discuss the incorporation of accepted human factors guidelines or principles into the design of displays for accident monitoring instrumentation. Therefore, the staff believes that identifying the NRC guidance documents that contain human factors guidelines or principles that are acceptable to the staff in RG 1.97, Revision 5, is appropriate. Consistent with RG 1.97, Revision 4, the staff will delete this staff position and will include the following statement in Section B of RG 1.97, Revision 5:

> Clauses 5.3, "Response time;" 8.1.2, "Human factors;" and 8.5, "Display location" discuss the incorporation of human factors guidelines or "accepted" human factors principles into the design of displays for accident monitoring instrumentation. The NRC provides additional guidance in Chapter 18, "Human Factors Engineering," of the NRC's Standard Review Plan (NUREG-0800), as well as in NUREG-0700, and NUREG-0711.

i. Editorial Comments

Comment [PWROG-5]: In page 3, NUREG-0800 first sub-bullet – remove the last word "appropriate".

NRC Response: The NRC agrees with this comment; however, the staff has decided to remove SRP Branch Technical Position 7-19, "Guidance for Evaluation of Diversity and Defense-in-Depth in Digital Computer-Based Instrumentation and Control Systems," from the "Related Guidance" section in RG 1.97 because it is no longer referenced.

Comment [PWROG-6]: In page 3, RG 1.152 bullet – remove the word "he" (the he most).

NRC Response: The comments are no longer applicable. The staff has decided to remove the related language and just list the related RGs which endorse normative references listed in Section 2.1 of IEEE Std. 497-2016.

Comment [DTE-2]: In page 3, add reference to IEEE Std. 279 along with the IEEE Std. 603 in the last bullet, based on the LDB of the operating plants.

NRC Response: The comments are no longer applicable. The staff has decided to remove the related language and just list the related RGs which endorse normative references listed in Section 2.1 of IEEE Std. 497-2016.

Comment [DTE-1]: In page 5, paragraph starting "Accordingly, IEE broadened the scope …" Replace "…IEE broadened the scope…" with "…IEEE broadened the scope…" and replace "…standard to include severe accidents." with "…standard to include consideration of the instrumentation potentially required for coping with severe accidents."

NRC Response: The NRC agrees with this comment and has revised the RG as appropriate.

Comment [DH-2]: Paragraph starting "Accordingly, IEE broadened the scope …" Replace "…IEE broadened the scope…" with "…IEEE broadened the scope…" and replace "…standard to include severe accidents." with "…standard to include consideration of the instrumentation potentially required for coping with severe accidents."

NRC Response: The NRC agrees with this comment and has revised the RG as appropriate.

Comment [PWROG-7]: In page 5, last paragraph – IEE should be IEEE.

NRC Response: The NRC agrees with this comment and has revised the RG as appropriate.

Comment [IEEE1-5]: In page 9, the term "a fuel damage" is rather unconventional English.

NRC Response: The NRC agrees with this comment and has revised the RG as appropriate.

Comment [IEEE1-8]: In page 10, Change IEEE Std. 379-2016 to IEEE Std. 379-2014. There is no 2016 version.

NRC Response: The NRC agrees with this comment; however, the staff has decided to remove IEEE Std. 379, "IEEE Standard for the Application of the Single-Failure Criterion to Nuclear Power Generating Station Safety Systems," because it is no longer referenced.

Comment [DH-9]: In page 11, correct "… change in the way the standard …" to "… change the way the standard …" for readability.

NRC Response: The NRC agrees with this comment and has revised the RG as appropriate.