Response to Public Comments on Draft Regulatory Guide (DG)-1380 Acceptability of ASME Code Section III, Division 5, "High Temperature Reactors" and NUREG-2245, "Technical Review of the 2017 Edition of ASME Section III, Division 5, `High Temperature Reactors'"

On August 20, 2021, the NRC published a notice in the *Federal Register* (86 FR 46888) that Draft Regulatory Guide, DG-1380 (Proposed Revision 2 of Regulatory Guide [RG] 1.87; Agencywide Documents Access and Management System [ADAMS] Accession No. ML21091A276) and NUREG-2245 (ADAMS Accession No. ML21223A097) were available for public comment. The Public Comment period ended on October 19, 2021. Subsequent to the public comment period for DG-1380, the NRC staff completed its review of Code Cases N-872 and N-898, related to the use of Nickel-Based Alloy 617. On March 1, 2022, the NRC staff issued a supplemental *Federal Register* notice (87 FR 11490) to DG-1380 requesting public comment on the staff's proposed endorsement of Code Cases N-872 and N-898. The public comment period for the supplemental *Federal Register* notice ended on March 31, 2022. The NRC received comments from the organizations listed below. The NRC has combined the comments and NRC staff responses in the following table.

Comments were received from the following:

Letter No.	ADAMS Accession No.	Commenter Affiliation	Commenter Name
1	ML21286A738	--	N. Prasad Kadambi
3	ML21292A289	Nuclear Energy Institute (NEI)	Mark A. Richter
4	ML21292A291	GE-Hitachi	Michael Arcaro

Comments on DG-1380:

Commenter	Specific Comments	NRC Staff Resolution
N. Prasad Kadambi	It is unfortunate that the NRC staff has chosen to use the old-fashioned guidance structure of updating a LWR Regulatory Guide supported by a NUREG for advanced reactor high-temperature components. It would have been more beneficial to use a structure that would be amenable to accomplish the aspirations of the Commission's direction in SRM-SECY-98-0144, "White Paper on Risk-Informed and Performance-Based Regulation" (White Paper). Such a structure would seek to accomplish outcomes consistent with	The staff disagrees with this comment. The comment implies that the NRC's endorsement of the ASME Section III, Division 5 code (Section III-5) is inconsistent with the use of risk-informed, performance-based approaches for licensing of advanced non-light water reactors (ANLWR). The staff's simple aim in revising RG 1.87, however, was to determine whether the

	recent statements by the staff regarding regulating toward "reasonable assurance of adequate protection". Clearly such a structure would focus on safety decision-making rather than specifying a process for compliance with a prescriptive set of rules that is codified in the ASME Section III, Division 5 (S-III-5) standard. The decision-making would take as input information from results produced by application of S-III-5 to a set of components. This set of components would be functional contributors to some significant feature of an advanced reactor design. The application of S-III-5 to the components would produce information which characterizes the capabilities of systems that support the design feature. The designer incorporates these system capabilities to achieve functional purposes to be provided by the design feature. The functional requirements associated with design feature would be met by systems that perform to set criteria to deliver physical needs of the design. Ideally, the functional requirements and criteria would be demonstrably fit-for-purpose with no unnecessary requirements. Achieving all this would be in keeping with the Commission's White Paper. The range of application for S-III-5 is vast when liquid-metal-cooled, gas-cooled, and molten-salt fueled or cooled reactors are considered. The multitude of possibilities of materials, construction methods, and service environments make sensible prescriptive approaches almost impossible. Yet the combination of NUREG-2235 and DG-1380 that the NRC staff has chosen to employ uses just such an approach. It appears that the regulated community is so much in need of S-III-5 that no negative comments have come forth so far even though the cost impacts are likely to be substantial and sub-optimal. It appears that this community is not paying attention to the costs of implementing S-III-5. Under the circumstances, it falls to the NRC to meet its obligations under the Nuclear Energy Innovation and Modernization Act (NEIMA) to find a risk-informed and performance-based (RIPB) approach to fulfill its role in reducing the costs of advanced reactors.	ASME BPV Code, Section III, Division 5, is an acceptable method for assuring the integrity of structures, systems, and components under specified conditions, including temperatures higher than those specified in Section III, Division 1. The staff is not mandating the use of ASME Boiler and Pressure Vessel Code, Section III, Division 5 (Section III-5, or S-III-5 in the terminology used in the comment). DG-1380 states at the beginning of Section C.1 that "[t]he NRC staff endorses the 2017 Edition of the ASME Code, Section III, Division 5, as a method acceptable to the NRC staff for the materials, mechanical/structural design, construction, testing, and quality assurance of mechanical systems and components and their supports of high-temperature reactors, with the exceptions and limitations stated below." The staff's endorsement with exceptions and limitations in DG-1380 of Section III-5 does not preclude the use of performance-based or risk-informed approaches. The comment states that "the range of application for S-III-5 is vast when liquid-metal-cooled, gas-cooled, and molten-salt fueled or cooled reactors are considered. The multitude of possibilities of materials, construction methods, and service environments make sensible prescriptive approaches almost impossible." The staff agrees that ANLWRs may have a wide range of coolants, materials and construction methods. Section III-5 provides methods to prevent certain failure modes, such as overload, stress rupture, creep and creep-fatigue, but does not address the effects of the coolant environment on materials. Aspects such as corrosion and irradiation will have to be addressed by applicants for

	S-III-5 requires the designer to provide a complete Design Specification which fulfills Owner/Operator responsibilities while also complying with whatever the local regulatory authority requires. The combination of NUREG-2235 and DG-1380 shows scant recognition of the fact that Design Specifications that draw only on S-III-5 would be quite incomplete. S-III-5 does look to ASME Section III, Division 1 rules for many needs. However, this only makes the process more prescriptive and convoluted. The pursuit of a less prescriptive approach needs to look to what the Commission has explicitly offered by way of remedies for this type of situation in the White Paper. It should be clear to the staff at this point of its rulemaking that the needs for the 10 CFR Part 53 would motivate the NRC staff to seek RIPB solutions for advanced reactors. RIPB solutions will require that the design function not operate in a silo, ignoring construction and operation needs as has been the practice in the past with the existing LWR fleet. The combination of NUREG-2235 and DG-1380 essentially continues existing practices by ignoring the Commission's recognition that NRC should offer flexibility to determine how to meet the established performance criteria in ways that will encourage and reward improved outcomes. In the context of 10 CFR Part 53, the improved outcomes clearly relate to functional success and not just avoidance of component failure. The NRC staff has immediate access to a number of guidance documents and research products that could address an RIPB approach to S-III-5. NUREG/BR-0303, "Guidance for Performance-Based Regulation" offers guidance for employing risk-informed, performance-based, and RIPB approaches to all NRC regulated activities. It also offers a methodology for using a structured set of performance objectives capable of dealing with the integrated decision-making necessary to roll-up component performance capabilities into functional performance success. Early research related to alternatives to prescriptive regulation is documented in	ANLWR designs by means other than those provided in Section III-5, such as environment-specific materials qualification programs, and in-situ surveillance programs during operation, or other strategies to provide reasonable assurance of component reliability. The NRC is also reviewing for endorsement ASME BPV Code Section XI, Division 2, "Reliability Integrity Management (Section XI-2)," which allows the use of diverse strategies to ensure component reliability. Section XI-2 includes risk-informed and performance based approaches, such as suggested by the comment. One of the strategies provided by Section XI-2 for reliability integrity management is design practices, so the design process for ANLWRs need not operate in a silo as implied by the comment. The requirement for the designer to provide a complete Design Specification as required by Section III-5 does not preclude designers from addressing other design aspects that are not addressed by Section III-5, and would not hinder the user of risk-informed and performance-based approaches to address regulatory requirements. No changes were made to the RG or NUREG-2245 as a result of this comment.

	NUREG/CR-5392, "Elements of an Approach to Performance-Based Regulatory Oversight". Additional research related to decision-making under uncertainty is documented in NUREG/CR-6833, "Formal Methods of Decision Analysis Applied to Prioritization of Research and Other Topics". The American Nuclear Society's Risk-Informed, Performance-Based Principles and Policy Committee (RP3C) has considerable information on application of RIPB methods for advanced reactors. In summary, the combination of NUREG-2235 and DG-1380 falls short of providing guidance that would accomplish the objectives of NEIMA. The NRC staff should recognize this as part of the NRC-2021-0177 comment resolution process. Additionally, there is an opportunity to propose activity that focuses on achieving the Commission's objectives in the White Paper.	
NEI	Should add a statement that Code Cases may be implemented upon ASME Committee approval.	The staff disagrees with this comment. It appears that NEI is suggesting that Code Cases should be automatically approved by NRC when the Code Cases are approved by the ASME Code Committee. The NRC must independently review code cases that have been approved by ASME to determine if the code cases would comport with NRC requirements prior to approving them for use. No changes were made to the RG or NUREG-2245 as a result of this comment.
NEI	Should add a statement that deviations from Code Case may be made with appropriate 50.59 analysis or equivalent analysis.	It is not clear from the comment whether the comment is intended to be restricted to Code Cases or to the Code, broadly. Nevertheless, the NRC staff agrees that in some instances 10 CFR 50.59, "Changes, tests and experiments," may be available to make changes of this sort. Because the NRC staff anticipates that applicants or licensees would incorporate this RG into their licensing basis in different ways, it is premature at this point for the

		NRC to establish whether 10 CFR 50.59 or other change control processes would be available or applicable. This RG is a guidance document, not a regulation. Applicants and licensees are always free to use other means to demonstrate that reactor designs meet the applicable regulations. No changes were made to the RG or NUREG-2245 as a result of this comment.
NEI	Section C.1.z(1); Extrapolation to determine the allowable time for use-fractions is an intended use of the Code, both to obtain t_{ib} in HGB-3224(d) and in other portions of the Code, including those referenced by the staff in the discussion of NUREG-2245 page 3-193. Extrapolation is not prohibited elsewhere in the Code; the Code is silent on extrapolation in the referenced paragraphs, which does not prohibit extrapolation as indicated in the Foreword to the Code, "the Code does not address all aspects of these activities and those aspects that are not specifically addressed should not be considered prohibited." Prohibiting extrapolation for determining allowable times may place an economic penalty on designs by restricting component design life or requiring significant overdesign to obtain the required life. It is noted that HGB-1124 restricts the time at elevated temperature to the maximum time associated with Smt; extrapolation does not permit increasing the operating time at elevated temperature beyond the restriction of HGB-1124, but rather allows for calculation of the use-fraction in conditions of low operating stress relative to the allowables. Restricting extrapolation for a component with a specified 300,000-hour design life at elevated temperature results in a use-fraction of greater than or equal to 1.0 regardless of the specified Service Loadings; this would occur because the denominator in the use-	The staff agrees with this comment, and has revised the RG to remove the limitation related to extrapolation in HGB-3224(d) and made conforming changes in the NUREG report.

	fraction summation would always be less than or equal to 300,000 hours. To achieve a time fraction of 1.0 in this case, all Service Level A, B, and C loadings would be required to have a stress less than or equal to S_t at 300,000 hours at the appropriate temperature, even if the Service Loading duration was much shorter, with higher stresses permitted by HGB-3224(c) equation (10). The most significant contributors to the use-fraction summation will be Service Loadings where the stresses are relatively high, and the allowable times have limited or no extrapolation. The Code margins for these Service Loadings are not at risk of being degraded by extrapolation. Lower stress Service Loadings, where t_{ib} is extrapolated to longer times, would be smaller overall contributions to the use-fraction summation since the total duration of all elevated temperature service loadings is limited to the time associated with S_{mt}. Since the low stress Service Loadings would have small overall contribution to the use-fraction, extrapolation error in these cases would not have a significant impact on the overall margins.	
NEI	Appendix A – General; There are numerous places within Appendix A that are inconsistent with 10 CFR 50.69. See comments below for specific examples of where Appendix A is inconsistent with 10 CFR 50.69.	The NRC staff agrees with this comment to the extent that an applicant or licensee need not comply with specified 10 CFR Part 50 requirements with respect to RISC-3 components in accordance with 10 CFR 50.69. In addition to changes made to address subsequent specific examples, the staff clarified that the definition of safety-related in 10 CFR 50.2 used in both traditional and 10 CFR 50.69 SSC classification processes may not be fully applicable to high temperature reactor designs. Further, Quality Group D was removed from the Appendix as it relates to NSR SSCs that are not important to safety and the designers and owners are responsible for assigning the appropriate standards for these SSCs.
NEI	Appendix A, A-2. Safety Classification Categories – Traditional Approach, Page A-2 (19 of 26);	The NRC staff agrees with the first part of this comment, but the staff disagrees with the proposed new paragraph. Accordingly, the staff significantly revised Appendix A,

	It is important to point out that in RG 1.26 Quality Group D is applied only to *"water- and steam-containing components that are not part of the reactor coolant pressure boundary or included in Quality Groups B or C but are part of systems or portions of systems that contain or may contain radioactive material."* The first two full paragraphs should be combined into one paragraph and re-written as shown below. Proposed New paragraph: SSCs that are NSR may function to prevent a radiological release to the public by ensuring that no dose to the public is beyond the regulatory limits of 0.1 rem total effective dose equivalent (TEDE) set by 10 CFR Part 20, "Domestic Licensing of Production and Utilization Facilities," (Ref. A-5). While such SSCs do not meet the criteria for an SR SSC, there is still a need to ensure component integrity. RG 1.26 assigns Quality Group D to components that contain or may contain radioactivity but are not part of the reactor coolant pressure boundary or included in Quality Groups B or C. Refer to RG 1.26 for more information on this traditional approach. RG 1.143, "Design Guidance for Radioactive Waste Management Systems, Structures, and Components Installed in Light-Water-Cooled Nuclear Power Plants," (Ref. A-10) provides information related to the classification of radioactive waste management systems that fall within the scope of that RG. SSCs that are NSR and do not meet the criteria for special treatment are left to the applicant to specify any standards for design and fabrication.	Section A-2. The staff reworded the second paragraph of Section A-2 to clarify the applicability of Quality Group D (as defined in RG 1.26) and RG 1.143. The staff also clarified that RG 1.26 and RG 1.143 endorsed standards for components within the scope of Quality Group D and the scope of RG 1.143. Finally, the staff stated that certain endorsed standards include high temperature operating conditions within their scope that may be appropriate for high temperature reactor applications, and the adequacy of these standards may be addressed during the review of an application for a specific design.
NEI	Appendix A, A-2. Safety Classification Categories – Traditional Approach, Page A-2 (19 of 26); Last full paragraph states*: "NSR mechanical components that need special treatment, such as for systems containing high levels of radioactive material…"*	The staff disagrees with this comment. There may be additional reasons that special treatment may be appropriate, not just because a system contains radioactive material. The RG was revised to provide an additional example, defense-in-depth, for the application of special treatment to NSR mechanical equipment. The staff also clarified the discussion of NSR SSCs.

	Change to: "NSR mechanical components that ~~need special treatment, such as for systems~~ contain~~ing~~ high levels of radioactive material…" as this part of section A-2 only applies to the "Traditional Approach" for Safety Classification Categories.	
NEI	Appendix A, A-2. Safety Classification Categories – Risk Informed Approach, Page A-3 (20 of 26); Second full paragraph is inconsistent with 10 CFR 50.69. Needs to be re-written so that for RISC-2 components the owner has the flexibility allowed by 10 CFR 50.69 and that for RISC-3 components, Section III and Appendix B are not required.	The staff agrees with this comment to the extent the draft language was confusing or could have been understood as overly restrictive. 10 CFR 50.69(d) requires, in part, that, if the risk-informed categorization process is voluntarily adopted by an applicant, the applicant shall ensure that RISC-1 and RISC-2 SSCs perform their safety-related functions consistent with the categorization process assumptions, and that the treatment of RISC-3 SSCs is consistent with the categorization process. The RG was revised to indicate standards that may be used with appropriate justification to demonstrate categorization process assumptions are satisfied for RISC-2 and RISC-3 components and to indicate that the NRC endorsed the ASME Code, Section III, Division 5 standard as an appropriate standard to meet regulatory requirements applicable to RISC-1 components.
NEI	Appendix A, A-4 Quality Group Classifications, Pages A-5 and A-6 (22 and 23 of 26); Should be re-written to be consistent with 10 CFR 50.69 (i.e., for Group B and C the owner defines these requirements). For Group B the owner also needs to provide "reasonable confidence." For Group C, the "requirements" need to be consistent with the categorization process.	The staff disagrees with this comment. Section 50.69 does not control the quality group for an SSC classified using the traditional or LMP approaches. The staff nonetheless revised the RG to clarify how to determine SSC quality group depending on the classifications that result from the three approaches (addressed in the Appendix) an applicant could take to SSC classification and the safety significance of the SSC's functions. The staff also revised the RG to indicate that the standards acceptable to the staff are based on staff judgement without full knowledge of the design details and that other standards and quality assurance aspects may be appropriate depending on the design details.

NEI	Appendix A, Table A-1, Page A-7 (24 of 26); Should be re-written to be consistent with 10 CFR 50.69 (i.e., for Group B and C the owner defines these requirements / applicable codes and standards). For Group B the owner also needs to provide "reasonable confidence." For Group C, the "requirements" need to be consistent with the categorization process.	The staff disagrees with this comment. Section 50.69 does not control the quality group for an SSC classified using the traditional or LMP approaches. The staff nonetheless revised the RG to clarify how to determine SSC quality group depending on the classifications that result from the three approaches (addressed in the Appendix) an applicant could take to SSC classification and the safety significance of the SSC's functions. The staff also revised the RG to indicate that the standards acceptable to the staff are based on staff judgement without full knowledge of the design details and that other standards and quality assurance aspects may be appropriate depending on the design details. Table A-1 is the staff's recommendation based on limited knowledge of the design for a reactor. The information in the table is guidance, not a requirement. The RG was revised to annotate Table A-1 to indicate that alternatives may be appropriate, and to clarify that alternative treatment under 10 CFR 50.69 for SSCs categorized as RISC-1, RISC-2, RISC-3, or RISC-4 requires NRC review and approval in accordance with 10 CFR 50.69.
NEI	Appendix A, Table A-1, Page A-7 (24 of 26); Table A-1 is not consistent with 10 CFR 50.69. The interpretation of Table A-1 is such that the user is required to use the codes and standards as defined in the table for the specified quality groups. However, there may be alternative design and construction codes Class A and B applicable and acceptable for Quality Group A, B, C and D components. Table A-1 should be re-written.	The staff does not agree with this comment. Table A-1 is the staff's recommendation based on limited knowledge of the design for a reactor. The information in the table is guidance, not a requirement. The staff nonetheless rewrote Table A-1 to clarify the guidance.

Comments on NUREG-2245:

Commenter	Specific Comments	NRC Staff Resolution
NEI	Page 3-107, lines 16-20; [NUREG-2245 Text]: *"The NRC staff is not endorsing Mandatory Appendix HBB-I-14 for: (a) Type 304 stainless steel (Type 304 SS) values of Smt, St, and Sr for temperatures greater than 1300 °F or 700 °C."* As the basis for the above restriction, NUREG-2245 Sections 3.7.5, 3.7.6, and 3.7.9 utilized comparisons in ANL/AMD-21/1, Tables 3 and 4. The staff proposed a cutoff at temperatures where the difference is -10% or greater. Review of Tables 3 and 4 of ANL/AMD-21/1 indicates that typically the difference does not reach -10% until longer times, for example at 725°C St in Table 3 does not drop below the 10% criteria until 100,000 hours. Has the staff considered use of both temperature and time to set this limit and allow short duration conditions at temperatures greater than 1300 °F or 700 °C, where the 2017 Code allowable stresses meet the 10% criteria?	The staff agrees with this comment, and notes that the same limitation is in DG-1380. After further review, the staff revised the limitations on S_{mt}, S_t, and S_r for Type 304 stainless steel in Section C.1.u(1)(a) of the RG, to be dependent on both temperature and time. Conforming changes were made in NUREG-2245, Section 3.7.
NEI	Page 3-107, lines 21-22; [NUREG-2245 Text]: *"The NRC staff is not endorsing Mandatory Appendix HBB-I-14 for: […] (b) Type 316 stainless steel (Type 316 SS) Sr values for temperatures greater than 1300 °F or 700 °C."* As the basis for the above restriction, NUREG-2245 Section 3.7.9 utilized comparisons in ANL/AMD-21/1 Table 6. The staff proposed a cutoff at temperatures where the difference is -10% or greater. Review of Tables 6 of ANL/AMD-21/1 indicates that typically the difference does not reach -10% until longer times, for example, at 725°C Sr in Table 6 does not drop below the 10% criteria until 1,000 hours. Has the staff considered use of both temperature and time to set this limit and allow short duration	The staff agrees with this comment, and notes that the same limitation is in DG-1380. After further review, the staff revised the limitation on the S_r values for Type 316 stainless steel in C.1.u(1)(b) of DG-1380 to be dependent on both temperature and time. Conforming changes have been made in NUREG-2245, Section 3.7.

	conditions at temperatures greater than 1300°F or 700°C, where the 2017 Code allowable stresses meet the 10% criteria?	
NEI	Page 3-107, lines 24-25; [NUREG-2245 Text]: *"The NRC staff is not endorsing Mandatory Appendix HBB-I-14 for: […] (c) 2-1/4Cr-1Mo material Smt, St, and Sr values for temperatures greater than 950 °F or 510°C.* As the basis for the above restriction, NUREG-2245 Sections 3.7.5, 3.7.6, and 3.7.9 utilized comparisons in ANL/AMD-21/1, Tables 10 and 11 and Figure 4. Review of Tables 10 and 11 indicates that the 2017 Code allowable stresses were conservative at 100,000 hours up to 550°C. Has the staff considered use of both temperature and time to set this limit and allow short duration conditions at temperatures greater than 950 °F or 510°C?	The staff agrees with this comment, and notes that the same limitation is in DG-1380. After further review, the staff revised the limitation on the S_{mt}, S_t, and S_r values for 2-1/4Cr-1Mo in C.1.u(1)(c) of the RG to be dependent on both temperature and time. Conforming changes have been made in NUREG-2245, Section 3.7.
GE-Hitachi	Abstract, Page iii, line 3; NUREG 2245 has evaluated 2017 edition of ASME III Division 5. New reactor designs will need to evaluate reactor design against applicable codes and standards in effect 6 months before the docketed date of the licensing application. There are over 20 changes (records) proposed against ASME III Division 5 2021 edition. Are there plans to update NUREG 2245 to current issued ASME code?	The comment seems to imply that newer versions of codes and standards available 6 months before the docketing of an application should be used, regardless of NRC's review and endorsement of the revised codes and standards. The NRC does not have such a requirement. An applicant or licensee that chooses to use RG 1.87, Revision 2 (final version of DG-1380), will use the editions of the Code endorsed in the RG. An applicant that chooses to use a version of the ASME Code, Section III, Division 5, that is not endorsed in RG 1.87, Revision 2, will need to justify any deviations from the provisions of the 2017 and 2019 Editions of the Code endorsed in RG 1.87, Rev. 2. There are currently no plans to update NUREG-2245. The NRC staff will likely consider the need to review later editions of ASME Section III, Division 5 for endorsement as needed and would consider a request by ASME and/or industry stakeholders to do so.

		No changes were made to the RG or NUREG-2245 as a result of this comment.
GE-Hitachi	Section 1.4, Pages 1-3 and 1-4; Section 1.4, Review of ASME Code Section III Division 5 and Associated Code Cases, page 1-4, line 5. The only current code cases in scope of NUREG 2245 are N-861 and N-862. There are numerous CC listed against ASME III D5 such as N-290-3, N-812-1, N-822-4, N-872, N-875, N-898. Is there a future plan to endorse the active code cases?	Since the issuance of DG-1380 the NRC staff completed its review of two additional Code Cases, namely N-872 and N-898. The staff's endorsement of these additional code cases is included in the final version of the RG 1.87 Revision 2. The review of other Section III, Division 5 code cases is out of the scope of the current effort, but the staff anticipates these code cases could be addressed in future revisions of RG 1.87. No changes were made to the RG or NUREG-2245 as a result of this comment.
GE-Hitachi	Section 3, Page 3-1 lines14-15; The assessments of NUREG 2245 reviews HBB sections with CC 1592, 1593, 1594, 1595, 1596. Can these legacy CC be provided as public documents on NRC webpage [Home Nuclear Reactors New Reactors Advanced Reactors (non-LWR designs) Endorsement Review of ASME B&PV Code Section III, Division 5, "High Temperature Reactors]?	The comment refers to documents that are copyrighted by ASME. Copies of these documents can be purchased from ASME, Two Park Avenue, New York, NY 10016-5990; telephone (800) 843-2763. Purchase information is available through the ASME Web-based store at https://www.asme.org/publications-submissions/publishing-information. No changes were made to the RG or NUREG-2245 as a result of this comment.
GE-Hitachi	Section 4, Pages 4-1 - 4-7; Section 4, Technical Review of Code Cases N-861 and N-862. Will these code cases be available as public documents in NRC webpage [Home Nuclear Reactors New Reactors Advanced Reactors (non-LWR designs) Endorsement Review of ASME B&PV Code Section III, Division 5, "High Temperature Reactors]?	The comment refers to documents that are copyrighted by ASME. Copies of these documents can be purchased from ASME, Two Park Avenue, New York, NY 10016-5990; telephone (800) 843-2763. Purchase information is available through the ASME Web-based store at https://www.asme.org/publications-submissions/publishing-information. No changes were made to the RG or NUREG-2245 as a result of this comment.
GE-Hitachi	Section 5, Pages 5-1 - 5-5;	The NRC staff will continue to actively participate in the Section III, Division 5 code committees. It is possible

	Section 5 of NUREG 2245-Draft Report for Comment lists exceptions and or limitations to ASME Section III Division 5 2017 edition. Is there future scope between NRC and ASME to update ASME III-D5 to disposition the NRC exceptions and limitations and re-assess future revision of ASME III Division 5 under revision to NUREG 2245?	that the NRC limitations and exceptions could be removed if (1) The ASME code committees revise Section III, Division 5 to address the NRC limitations and exceptions, or (2) provide additional technical bases supporting the existing code provisions on which the NRC is imposing limitations. No changes were made to the RG or NUREG-2245 as a result of this comment.