August 28, 2014

MEMORANDUM TO: Stacey L. Rosenberg, Chief
Vessels and Internals Integrity Branch
Division of Engineering
Office of Nuclear Reactor Regulation

FROM: Gary L. Stevens, Sr. Materials Engineer ***/RA/***
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SUBJECT: RESPONSE TO PUBLIC COMMENTS ON DRAFT NRC REGULATORY ISSUE SUMMARY 2014-XX INFORMATION ON LICENSING APPLICATIONS FOR FRACTURE TOUGHNESS REQUIREMENTS FOR FERRITIC REACTOR COOLANT PRESSURE BOUNDARY COMPONENTS

A notice of opportunity for public comment on the subject regulatory issue summary was published in the *Federal Register* (79 FR 21812) on April 17, 2014. Comments were received from the Electric Power Research Institute (ML14141A414) and the Pressurized Water Reactor Owners Group (ML14141A413). Enclosed are the staff responses to all public comments.

Enclosure:
As stated

CONTACT: Gary L. Stevens, RES/DE/CIB
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ADAMS Accession No.: ML14192B402 *concurred via e-mail **NRR-106**

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Analysis of Public Comments on
DRAFT NRC REGULATORY ISSUE SUMMARY 2014-XX
INFORMATION ON LICENSING APPLICATIONS FOR FRACTURE TOUGHNESS REQUIREMENTS FOR FERRITIC REACTOR COOLANT PRESSURE BOUNDARY COMPONENTS (ML13301A188)

Comments on the subject draft regulatory issue summary are available electronically at the U.S. Nuclear Regulatory Commission's (NRC's) electronic Reading Room at http://www.nrc.gov/reading-rm/adams.html. From this page, the public can gain entry into Agencywide Documents Access and Management System (ADAMS), which provides text and image files of NRC's public documents. Comments were received from the following individuals or groups:

Letter No.	ADAMS Accession No.	Commenter Affiliation	Commenter Name	Abbreviation
1	ML14141A414	Electric Power Research Institute	Tim Hardin	EPRI
2	ML14141A413	Pressurized Water Reactor Owners Group	Jim Molkenthin	PWROG

This document lists each public comment by Letter No. For each comment, the NRC has either repeated the comment as written by the commenter or summarized the comment for conciseness and clarity. Each comment is referred to in the form [XXX]-[YYY]-[ZZZ], where: [XXX] represents the Abbreviation from the above table, [YYY] represents the Letter No. from the above table, and [ZZZ] represents the sequential comment number from that commenter.

EPRI-1-1 Comment

[Regarding "Background Information," third sentence: "Reactor vessel material toughness is monitored using P-T limits and Charpy upper shelf energy."]

Reactor vessel material toughness is not "monitored" using P-T curves. Reactor vessel material toughness is monitored by the reactor vessel material surveillance program required by 10CFR50 [Part 50 of Title 10 of the Code of Federal Regulations] Appendix H. The results of the surveillance program are used in the determination of the P-T curves. The purpose of the P-T curves is to limit vessel pressure at any given temperature, not to monitor toughness. Recommend revising this sentence as follows: "Reactor vessel material toughness is monitored using Charpy transition temperature shift and Charpy upper shelf energy."

NRC Response

NRC agrees with the comment. However, the intent of this regulatory issue summary (RIS) is to clarify requirements for P-T limits, not Charpy transition temperature and Charpy upper shelf energy. Therefore, rather than the suggested revision, which centers on Charpy transition temperature and Charpy upper shelf energy, the second and third sentences under "Background Information" will be revised to focus on P-T

limits: “Therefore, NRC regulations require monitoring of reactor vessel material fracture toughness during plant operation as stated in 10 CFR Part 50, Appendix H, ‘Reactor Vessel Material Surveillance Program Requirements.’ P-T limits are used to restrict reactor operation so that reactor vessel material toughness margins are maintained based on the results of the 10 CFR Part 50, Appendix H reactor vessel material surveillance program.”

In addition, the discussion of NRC Generic Letter 96-03 will be relocated to the end of the Background section of the RIS, with one additional sentence added afterward because the NRC considers the previous placement of this discussion to be out of place and context given the others revisions made to the RIS. This is reflected by the following change to the RIS:

> Finally, NRC Generic Letter 96-03, “Relocation of the Pressure Temperature Limit Curves and Low Temperature Overpressure Protection System Limits,” dated January 31, 1996, advises licensees that they may request a license amendment to relocate the P-T limit curves from their plant technical specifications (TS) to a pressure temperature limits report (PTLR) or a similar document, and states the following:
>
> > The methodology used to determine the P/T and LTOP [low temperature overpressure protection] system limit parameters must comply with the specific requirements of Appendices G and H to Part 50 of Title 10 of the *Code of Federal Regulations* (10 CFR), be documented in an NRC-approved topical report or in a plant-specific submittal, and be incorporated by reference into the TS. Subsequent changes in the methodology must be approved by a license amendment; 10 CFR 50.59 does not apply.
>
> Therefore, because PTLRs also document 10 CFR Part 50, Appendix G P-T limits, the clarifications made in this RIS also apply to those documents.

<u>EPRI-1-2a Comment</u>

[Regarding page 2 of 5, last paragraph, beginning “10 CFR Part 50, Appendix H,.....”, last sentence, “Therefore, the fracture toughness requirements of 10 CFR Part 50 Appendix G for the reactor vessel beltline are applicable to the reactor vessel materials with projected neutron fluence values greater than 1 x 10^{17} n/cm^2 (E > 1 MeV) at the end of the operating period.”]

*This statement is inaccurate. The fracture toughness requirements of 10 CFR Part 50 Appendix G are applicable to all ferritic components of the reactor coolant pressure boundary; fluence is not a criteria. Therefore the fracture toughness requirements of 10 CFR Part 50 Appendix G would be applicable to the reactor vessel beltline even if the projected neutron fluence were less than 1 x 10^{17} n/cm^2 (E > **1** MeV) at the end of the operating period.*

NRC Response

NRC agrees with the comment. In response to the comment, the last sentence of the last paragraph on page two will be revised to, “Therefore, the beltline definition in 10 CFR Part 50, Appendix G is applicable to all reactor coolant pressure boundary ferritic materials with projected neutron fluence values greater than 1 x 10^{17} n/cm^2 (E > 1 MeV) at the end of the operating period.”

In addition, “operating period” will be changed to “the design life” in the second sentence of this same paragraph to be consistent with the wording in 10 CFR Part 50, Appendix H.

EPRI 1-2b Comment

[Regarding page 2 of 5, last paragraph, beginning “10 CFR Part 50, Appendix H,.....”, last sentence, “Therefore, the fracture toughness requirements of 10 CFR Part 50 Appendix G for the reactor vessel beltline are applicable to the reactor vessel materials with projected neutron fluence values greater than 1 x 10^{17} n/cm^2 (E > 1 MeV) at the end of the operating period.”]

The paragraph that culminates with this statement attempts to establish that the regulations (10 CFR 50 Appendices G and H) define the beltline as materials with projected neutron fluence values greater than 1 x 10^{17} n/cm^2 (E > 1 MeV) at the end of the operating period. The logic used to reach this conclusion is not supported by the regulations. The fact that 10 CFR 50 Appendix H does not require a surveillance program for vessels in which the peak neutron fluence at the end of the design life of the vessel will not exceed 1017 n/cm2 (E > 1 MeV) does not constitute a definition of beltline; it is simply the threshold criteria for requiring a surveillance program. 10 CFR 50 Appendix H does not define beltline at all; instead, Appendix H Para. II, "Definitions", states, "All terms used in this Appendix have the same meaning as in Appendix G." Therefore, Appendix H adopts the same definition of beltline as Appendix G, which is,

> *"Beltline or Beltline region of reactor vessel means the region of the reactor vessel (shell material including welds, heat affected zones, and plates or forgings) that directly surrounds the effective height of the active core and adjacent regions of the reactor vessel that are predicted to experience sufficient neutron radiation damage to be considered in the selection of the most limiting material with regard to radiation damage."*

There is no stipulation of fluence level in that definition. RECOMMENDATION: Change the last sentence on page 2 of 5 to the following: "Therefore, the NRC Staff have traditionally interpreted the regulations as implying that the fracture toughness requirements of 10 CFR Part 50 Appendix G for the reactor vessel beltline are applicable to the reactor vessel materials with projected neutron fluence values greater than 1 x 10^{17} n/cm^2 (E > 1 MeV) at the end of the operating period."

NRC Response

NRC disagrees with the comment. Recent investigation of power reactor surveillance data by the NRC staff continues to support the use of a limit of $1x10^{17}$ n/cm^2 because that is the fluence level above which measured shifts in material properties become significant. These updated observations remain consistent with past NRC conclusions, which have been documented in past NRC reports. For example, support of the referenced statement in the draft RIS is given in Section 2.2, "Reactor Vessel Design," of NUREG-1511, which states:

> The beltline of the reactor vessel is defined in Appendix G, 10 CFR Part 50, as the region of the reactor vessel that directly surrounds the effective height of the active core and the adjacent regions of the reactor vessel that are predicted to experience sufficient neutron damage to be considered in the selection of the limiting material with regard to radiation damage. The NRC staff considered materials with a projected neutron fluence of greater than 1.0E17 neutrons per square centimeter (n/cm^2) at end of license (EOL) to experience sufficient neutron damage to be included in the beltline. This neutron fluence is based on the surveillance requirements in Appendix H, 10 CFR Part 50, and Figure 1 in Regulatory Guide 1.99, Revision 2.

Based on the foregoing, the NRC will make the following two changes to the RIS to further explain the fluence limit:

First, a new sentence will be added prior to the last sentence in the last paragraph on page 2 as follows, "...appendix H of this part. Furthermore, Section 2.2 of NUREG-1511, 'Reactor Pressure Vessel Status Report,' dated December 1994, states, 'The NRC staff considered materials with a projected neutron fluence of greater than 1.0E17 neutrons per square centimeter (n/cm^2) at end of license (EOL) to experience sufficient neutron damage to be included in the beltline. Therefore, the...' [refer to the new sentence added in response to Comment EPRI-1-2a for the remainder of this change].

Second, the second and third sentences of the first paragraph on page three will be revised as follows, "This threshold neutron fluence level is therefore reflected in the surveillance program criteria set forth in 10 CFR Part 50, Appendix H, as described in NUREG-1511. As a result of changes made to the plant license, including power uprates, increased operating periods due to license renewal, modified fuel design, neutron fluence methodology updates, and fuel placement within the core, the physical region of the reactor vessel with fluence that exceeds this level can expand."

EPRI-1-3 Comment

[Regarding page 3 of 5, third paragraph, second sentence, "Nevertheless, addressees must still be able to demonstrate that the P-T limits in license amendment requests and PTLRs developed for the plant do, in fact, bound all ferritic components of the reactor coolant pressure boundary as required by section I of 10 CFR Part 50, Appendix G."]

Here and throughout the RIS, the assertion is made that 10 CFR 50 Appendix G requires P-T limits to bound all ferritic components of the reactor coolant pressure boundary, but 10 CFR 50 Appendix G establishes no such requirement. Therefore, these statements, where they appear throughout the RIS, should be revised.

As noted in the RIS, 10 CFR 50 Appendix G, Section I, states that the appendix "...specifies fracture toughness requirements for ferritic materials of pressure retaining components of the reactor coolant pressure boundary of light water nuclear power reactors to provide adequate margins of safety during any condition of normal operation, including anticipated operational occurrences and system hydrostatic tests, to which the pressure boundary may be subjected over its service lifetime." Section I does not define what the fracture toughness requirements are; those are defined in Section IV, Fracture Toughness Requirements, which states, "The pressure-retaining components of the reactor coolant pressure boundary that are made of ferritic materials must meet the requirements of the ASME Code, supplemented by the additional requirements set forth below, for fracture toughness during system hydrostatic tests and any condition of normal operation, including anticipated operational occurrences." Therefore, all pressure-retaining components of the reactor coolant pressure boundary (RCPB) must (1) meet the requirements of the ASME Code and (2) the additional requirements below.

Those "additional requirements set forth below" are:

1. *Reactor Vessel Charpy Upper-Shelf Energy Requirements, and*
2. *Pressure-Temperature Limits and Minimum Temperature Limits.*

It should be noted that the ASME Code does not require P-T limits for the entire ferritic RCPB. Therefore, for the RIS to be correct, the requirement for P-T curves to bound the entire ferritic RCPB must be embedded in one of the two additional requirements. However, both of the additional requirements listed above are specific requirements for the pressure vessel. The Charpy upper-shelf energy requirements are specific for the vessel beltline materials. The scope of the second requirement (P-T limits and minimum temperature requirements) is likewise established in the first sentence IV.A.2.a: "Pressure-temperature limits and minimum temperature requirements for the reactor vessel are given in table 1, and are defined by the operating condition (i.e., hydrostatic pressure and leak tests, or normal operation including anticipated operational occurrences), the vessel pressure, whether or not fuel is in the vessel, and whether the core is critical." Table 1 is entitled "Pressure and Temperature Requirements ***for the Reactor Vessel****" [emphasis added]. All discussion in 10 CFR 50 Appendix G, IV.A.2, "Pressure-Temperature Limits and Minimum Temperature Requirements" is specific to the vessel; there exists no statement extending those requirements to the entire ferritic RCPB.*

There is no basis in regulation for the RIS statements that P-T limits must bound all ferritic components of the RCPB. Saying that Section I of 10 CFR 50 Appendix G extends the requirement for P-T to the entire ferritic RCPB is as inaccurate as saying that Section I extends the requirement for minimum 102 J Charpy upper shelf energy to the entire ferritic RCPB.

NRC Response

NRC agrees with the comment. In response to the comment, the third paragraph on page three will be deleted, and the following sentence will be added to the end of the previous paragraph, “In addition, all addressees should consider the effects of any replaced ferritic reactor coolant pressure boundary components on the adequacy of the P-T limits.”

Other changes to the RIS in response to this comment are as follows:
The last sentence of the second paragraph on page three will be revised, “Thus, the development of P-T limits for the reactor vessel must consider not only the vessel shell material with the highest reference temperature but also other vessel materials with structural discontinuities.”

The last paragraph on page three will be revised as described in the response to Comment EPRI-1-5a.

The “Summary of Issue” section on page four will be revised as described in the response to Comment EPRI-1-6.

Finally, refer to the response to Comment PWROG-2-10 for further changes that address this comment.

EPRI 1-4 Comment

[Regarding page 3 of 5, last paragraph]

The RIS states the expectation that components other than the reactor vessel should be accounted for in P-T curves but provides insufficient insight on how this is demonstrated for ferritic components outside the reactor vessel. The example (ML120820510) cited in the RIS does not provide any insight for how a plant which has ferritic piping components can satisfy the perceived requirements; nor, more specifically, how a plant which has ferritic piping components that were designed, specified, or fabricated prior to the adoption of the ASME III and 10 CFR 50 Appendix G rules introduced in 1972-73 can satisfy the perceived requirement.

NRC Response

This comment was addressed by the response to Comment EPRI-1-5a.

EPRI 1-5a Comment

[Regarding page 3 of 5, last paragraph, last sentence: “Such responses are consistent with licensees' understanding that has been communicated to NRC in the past (e.g., refer to page 5-8 of EPRI Report No. NP-5172-SR, Revision 1, “Primer: Fracture Mechanics in the Nuclear Power Industry,” May 1991).”]

The purpose of the reference to the EPRI report, page 5-8, is unclear. Page 5-8 discusses P-T limits for the vessel only and does not appear to support the contention in the RIS that 10 CFR 50 Appendix G P-T limits apply to the entire ferritic RCPB.

NRC Response

The intent of the reference to the EPRI report is to reinforce that licensees already have indicated they understand the intent of the RIS. As an example to demonstrate this, reference is made to the paragraph on page 5-8 of the EPRI report, which states:

> Although the reactor vessel beltline region is usually the most limiting location, other areas must also be considered in the determination of pressure-temperature limits. The reactor vessel flange or nozzle region may be limiting in some cases, particularly for newer plants and plants with low irradiation levels. If this is the case, a composite curve is constructed using the lowest, most conservative, limit for pressure as a function of temperature.

The above paragraph duplicates the intent of the RIS in that it states that all ferritic materials of the reactor vessel must be addressed, not just the beltline materials that experience embrittlement effects due to radiation. Based on this comment, the above report excerpt, and the responses to other comments, the last paragraph on page three will be revised to the following:

> During recent license amendment submittals by licensees pertaining to P-T limits, the NRC staff requested additional information from the applicants due to a lack of sufficient demonstration that all ferritic materials of the reactor vessel were addressed. These requests for additional information are consistent with NRC guidelines specified in NUREG-0800, Revision 2, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition," Section 5.3.2, "Pressure-Temperature Limits, Upper-Shelf Energy, and Pressurized Thermal Shock." To date, licensee responses to NRC requests for additional information have indicated that revised P-T limits have bounded all ferritic materials of the reactor vessel for the approved P-T limits periods. Such responses are consistent with licensees' understanding that has been communicated to NRC in...

This change also addresses Comment EPRI-1-4.

EPRI 1-5b Comment

[Regarding page 3 of 5, last paragraph, last sentence: "Such responses are consistent with licensees' understanding that has been communicated to NRC in the past (e.g., refer to page 5-8 of EPRI Report No. NP-5172-SR, Revision 1, "Primer: Fracture Mechanics in the Nuclear Power Industry," May 1991)."]

The EPRI report number is incorrect. The correct report number which will locate the report on the EPRI website is NP-5792-SRR1.

NRC Response

NRC agrees with the comment. In response to this comment, the parenthetical reference at the top of page four will be revised as follows, "(e.g., page 5-8 of EPRI Report No. NP-5792-SRR1, "Primer: Fracture Mechanics in the Nuclear Power Industry: Revision 1," dated May 1991)."

EPRI 1-6 Comment

[On page 4 of 5: "Summary of Issue", first sentence: "Some recent licensee submittals pertaining to ***reactor vessel P-T limits*** *[emphasis added] have lacked sufficient demonstration that all ferritic materials of the reactor coolant pressure boundary were addressed in accordance with the requirements of 10 CFR Part 50 Appendix G."]*

There is no requirement in 10 CFR 50 Appendix G or ASME Section XI Appendix G that P-T limits for the reactor vessel must bound all ferritic components of the reactor coolant pressure boundary (RCPB). Neither 10 CFR 50 Appendix G nor the RIS provide any guidance on how a licensee can address the Staff's desire that P-T limits address the entire ferritic RCPB. These observations suggest that major revisions to this draft RIS are necessary.

NRC Response

NRC agrees with the comment. In response to the comment, the "Summary of Issue" section on page four will be revised as follows:

> Some recent licensee submittals pertaining to reactor vessel P-T limits have lacked sufficient demonstration that all ferritic materials of the reactor vessel were addressed in accordance with the requirements of 10 CFR Part 50, Appendix G. Specifically, some submittals regarding P-T limits provided technical bases analyzing only the reactor vessel material with the highest reference temperature without supporting details to demonstrate that the resulting P-T limits were bounding for all ferritic components of the reactor vessel. In determining P-T limits, reactor vessel materials with the highest reference temperature may not always produce the most limiting P-T limits because the consideration of stress levels from structural discontinuities (such as nozzles) may produce a lower allowable pressure. All addressees should ensure that P-T limits (including those implementing NRC-approved PTLR methodologies) sufficiently address all ferritic materials of the reactor vessel, including the impact of structural discontinuities, and address the impact of neutron fluence accumulation in accordance with the requirements of 10 CFR Part 50, Appendix G. Specifically, all ferritic components within the entire reactor vessel must be considered in the development of P-T limits, and the effects of neutron radiation must be considered for any locations that are predicted to experience a neutron fluence exposure greater than $1x10^{17}$ n/cm^2 (E > 1 MeV) at the end of the licensed operating period.

PWROG 2-1 Comment

[Regarding page 1; 'Background Information'; 1st paragraph; 4th line (3rd sentence)] Reactor vessel toughness is not monitored using the P-T limits, it is monitored via the reactor vessel material surveillance program.

Please revise this sentence as follows:

"Reactor vessel material toughness is monitored via the reactor vessel material surveillance program~~using P-T limits and Charpy upper shelf energy~~."

NRC Response

This comment was addressed by the response to Comment EPRI-1-1.

PWROG 2-2 Comment

[Regarding page 1; 'Background Information'; 1st paragraph; last sentence]

The last sentence is not pertinent to this RIS and is not completely accurate. This RIS does not discuss Charpy upper shelf energy and does not need to define it. Some materials can lose significant fracture toughness with irradiation, but most do not.

Please delete the following sentence:

~~"Charpy upper shelf energy is a measure of the average energy absorbed by materials at a temperature that is above the upper end of the temperature transition region where materials lose significant fracture toughness."~~

NRC Response

NRC agrees with the comment. The last sentence of the first paragraph of "Background Information" will be deleted as indicated in the comment. In addition, the first paragraph after the italicized indent on page 2 of the RIS will be changed to, "In particular, 10 CFR Part 50, Appendix G specifies P-T limits and minimum temperature requirements for the reactor vessel."

PWROG 2-3 Comment

[Regarding page 2; last paragraph; 3rd sentence]

10 CFR 50, Appendix G, Section II.D.(ii) states that for beltline materials, "RT_{NDT} must account for the effects of neutron radiation." Under Section II.F. The beltline is clearly defined as materials "that directly surrounds the effective height of the active core and adjacent regions of the reactor vessel that are predicted to experience sufficient neutron radiation damage to be considered in the selection of the most limiting material with regard to radiation damage."

Please revise this sentence as follows:

"It is the staff's interpretation based on 10 CFR 50 Appendix H (and current understanding of irradiation embrittlement), that 1 x 10^{17} n/cm^2 (E > 1 MeV) is considered sufficient fluence to cause sufficient neutron radiation damage such that these materials should be considered in the selection of the most limiting material."~~Appendix G to 10 CFR Part 50 states, "To demonstrate compliance with the fracture toughness requirements of section IV of this appendix, ferritic materials must be tested in accordance with the ASIVE Code and, for the beltline materials, the test requirements of appendix H of this part.~~"

NRC Response

This comment was addressed by the response to Comment EPRI-1-2b.

PWROG 2-4 Comment
[Regarding page 2; last paragraph; last sentence]

Fracture toughness requirements of 10CFR50, Appendix G apply to the entire reactor vessel.

Please revise the last sentence as follows:

"Therefore, changes in the fracture toughness properties due to neutron irradiation should be considered for ~~requirements of 10 CFR Part 50 Appendix G for the reactor vessel beltline are applicable to~~ the reactor vessel materials with projected neutron fluence values greater than 1 x 10^{17} n/cm^2 (E > 1 MeV) at the end of the operating period."

NRC Response

This comment was addressed by the response to Comment EPRI-1-2a.

PWROG 2-5 Comment

[Regarding page 3; first paragraph; last sentence]

PWR nozzles are generally not positioned immediately above the active core height. A portion of the upper shell is between the nozzles and the beltline.

Please revise the last sentence as follows:

"In particular, this may be true for reactor vessel nozzle materials when the nozzles are positioned ~~immediately~~ above or below the active core height."

NRC Response

NRC agrees with the comment. The last sentence of the first paragraph on page three will be revised as indicated in the comment.

PWROG 2-6 Comment

[Regarding page 3; 2nd paragraph; 2nd sentence]

This sentence is confusing, please reword. It seems to say the opposite of this intent.

Please revise the sentence as follows:

"In the development of the P-T limits, it is not sufficient to only consider the limiting reactor vessel material (generally considered to be the vessel shell materials with the highest reference temperature), and not consider the stress levels due to structural discontinuities.~~The inclusion of stress levels from structural discontinuities makes consideration of only the limiting material for a particular reactor vessel (generally considered to be the vessel shell material with the highest reference temperature) in the determination of the P-T limits insufficient.~~"

NRC Response

NRC agrees with the comment. The second and third sentences of the second paragraph on page three will be revised to, "In the development of the P-T limits, it is not sufficient to only consider the limiting reactor vessel material (generally considered to be the vessel shell materials with the highest reference temperature) and not consider the stress levels due to structural discontinuities. This is because the effects of structural discontinuities for a material with a lower reference temperature (such as a nozzle with a lower neutron fluence) may result in more restrictive allowable P-T limits than those for the vessel shell material limiting material with a higher reference temperature."

PWROG 2-7 Comment

[Regarding page 3; 3rd paragraph]

The first sentence only provides an example which is adequately addressed in the previous paragraph. The second and third sentences should be clarified as proposed.

10 CFR 50, Appendix G, IV. A. states "The pressure retaining components of the reactor coolant pressure boundary that are made of ferritic materials must meet the requirements of the ASME Code, ... "

Additionally, as stated in 10 CFR 50, Appendix G, I. the second paragraph "If no section (of ASME Code) is specified, the reference is to Section III, Division 1,..."

Therefore, the ferritic RCPB components outside the reactor vessel are governed by ASME Section II1.

10 CFR 50, Appendix G does not impose ASME Section XI, Appendix G requirements on ferritic RCPB components outside the reactor vessel.

The second sentence should be revised to clarify that the RIS should be considered when the P-T limits are revised via a license amendment request (LAR), or when the P-T limits that are contained in the PTLR are revised. Please see the proposed revision to this paragraph.

Please replace the entire paragraph with the following:

"Addressees must be able to confirm that revised P-T limit license amendment requests and PTLR revisions consider all ferritic components of the RCPB as required by 10 CFR Part 50. Appendix G. Ferritic RCPB components outside of the reactor vessel that met all the applicable ASME Code, Section III requirements at the time that the RCPB components were constructed demonstrates compliance with 10 CFR Part 50. Appendix G and do not need any further consideration for the P-T limits. Replaced ferritic RCPB components, e.g., steam generators, must also meet the ASME Code, Section III requirements in accordance with 10 CFR Part 50, Appendix G and also do not need any further consideration for the P-T limits.~~Vessel materials with stress discontinuities and a lower reference temperature are unlikely to provide the lowest allowable P-T limits for a vessel except early in the plant operating life or in cases with limited differences in the reference temperatures for the vessel shell materials and the materials with stress discontinuities (for example where the vessel shell materials have low levels of neutron embrittlement). Nevertheless, addressees must still be able to demonstrate that the P-T limits in license amendment requests and PTLRs developed for the plant do, in fact, bound all ferritic components of the reactor coolant pressure boundary as required by section I of 10 CFR Part 50, Appendix G. In addition, this demonstration would need to consider the effects of any replaced ferritic reactor coolant pressure boundary components on the adequacy of the P-T- limits.~~"

NRC Response

This comment was addressed by the responses to Comments EPRI-1-6, PWROG-2-8, and PWROG-2-10.

PWROG 2-8 Comment

[Regarding page 4; 'Summary of Issue'; 4th sentence]

The RIS should only apply to future revisions of the P-T limits that are revised via a LAR, or when the P-T limits that are contained in the PTLR are revised. Please see the proposed revision to the 4th sentence in this paragraph.

10 CFR 50, Appendix G does not impose ASME Section XI, Appendix G requirements on ferritic RCPB components outside the reactor vessel.

Please revise the sentence as follows:

All addressees should ensure that revised P-T limits in license amendment requests and PTLR revisions ~~(including NRC-approved PTLRs)~~ sufficiently address all ferritic materials of pressure-retaining components of the reactor coolant ~~pressure boundary~~vessel, including the impact of structural discontinuities, and address the impact of neutron fluence accumulation in accordance with the requirements of 10 CFR Part 50 Appendix G.

NRC Response

NRC does not agree with the first part of the comment regarding future revision of P-T limits. As stated in the "Background" and "Summary of Issue" sections of the RIS, and further described in the responses to other comments (e.g., refer to the response to EPRI-1-6), 10 CFR Part 50, Appendix G requires that P-T limits address all ferritic portions of the reactor vessel. Therefore, if a licensee's P-T curves do not address all ferritic portions of the reactor vessel, revised curves should be developed to demonstrate compliance with the requirements of 10 CFR Part 50, Appendix G and submitted to the NRC for review and approval.

NRC agrees with the second part of the comment regarding the imposition of ASME Section XI, Appendix G requirements. This portion of the comment was addressed in the response to Comment EPRI-1-6.

PWROG 2-9 Comment

[Regarding page 4; 'Summary of Issue'; last sentence]

10 CFR 50, Appendix G does not impose ASME Section XI, Appendix G requirements on ferritic RCPB components outside the reactor vessel.

P-T curves are not tied to a license period.

Please revising the sentence as follows:
"Specifically, all ferritic components within the ~~entire RCPB~~reactor vessel must be considered in the development of P-T limits, and the effects of neutron radiation must be considered for any materials that are predicted to experience ~~an end of license~~ neutron fluence exposure greater than 1 x 10^{17} n/cm^2 (E > 1 MeV) at the end of the operating period."

NRC Response

This comment was addressed by the response to Comment EPRI-1-6.

PWROG 2-10 Comment

[Regarding page 4; 'Summary of Issue'; last sentence]

See comment 7.

Please add the following sentence:

"Additionally, ferritic RCPB components outside of the sentence reactor vessel that met all the applicable ASME Code, Section III requirements at the time that the RCPB components were constructed demonstrates compliance with 10 CFR Part 50, Appendix G and do not need any further consideration for the P-T limits."

NRC Response

NRC agrees with the comment. In addition, the NRC will add further explanation to identify where NRC guidance may be found to address the subject of this comment. Therefore, the NRC will add the following new paragraph to the end of the "Summary of Issue" section:

> As specified in Sections I and IV.A of 10 CFR Part 50, Appendix G, ferritic RCPB components outside of the reactor vessel must meet the applicable requirements of ASME Code, Section III, "Rules for Construction of Nuclear Facility Components". Further guidance on these requirements may be found in NUREG-0800, Revision 2, Section 5.3.2, Paragraph II, "Acceptance Criteria." Specifically, Items 1 through 4 of the Technical Rationale portion of that Paragraph discuss requirements for RCPB components.