NUMARC

NUCLEAR MANAGEMENT AND RESOURCES COUNCIL

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File

February 3, 1992

Mr. John Craig, Director
License Renewal Project Directorate
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Mail Code 13 D1
Washington, DC 20555

Dear Mr. Craig:

Enclosed is the PWR INTERNALS IR: RESPONSES TO NRC COMMENTS, dated 1/29/92. This comment response document has been developed to respond to the NRC's comments on the PWR Internals Industry Report (NUMARC Report Number 90-05) which were submitted to NUMARC on January 31, 1991.

These responses are submitted for the NRC's review prior to an anticipated meeting between the NRC staff and NUMARC. The purpose of the proposed meeting will be to develop a mutual understanding of the industry's responses to the NRC's comments. It is our goal to resolve as many as possible of the staff's comments during this meeting and to set the stage for a quick resolution of any outstanding issues after the meeting.

Currently, a date for this proposed meeting has not been established. A date in the first half of March 1992 is suggested. NUMARC will work with the NRC to firm up the actual meeting date.

Should you have any questions please contact me or Kurt Cozens of the NUMARC staff.

Sincerely,

Edward P. Griffing

Edward P. Griffing
Manager, Technical Division

KOC\
Enclosure

cc: Warren Minners, NRC/RES/DSIR
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NRC Document Control Desk
John DeVincentis, Chairman NUMARC NUPLEX Working Group

PWR INTERNALS IR: RESPONSE TO NRC COMMENTS

NRC COMMENT #	NRC COMMENT	INDUSTRY RESPONSE	STATUS	PROPOSED IR REVISION	REMARKS
G-1	The arguments for license renewal presented in the IR are based in large part on the existence of on-site inspection, maintenance and replacement programs. However, existence of these programs does not necessarily provide adequate grounds for life extension. The Individual Plant Assessment (IPA) includes an evaluation of programs to determine effectiveness.	The evaluation of potentially significant age-related degradation mechanisms in Section 5 of the IR depends on the determination that various aging management programs are established and effective, as pointed out by the reviewer. A license renewal applicant wishing to take credit for a generic conclusion in Section 5 is responsible for assuring that the referenced aging management program elements are in place. The NRC is responsible for enforcement and audit functions. It is the combination of IR findings, applicant affirmations,and NRC audit/enforcement that permits the applicant's license to be renewed. There is no conflict among the three.	Open	Standardized language revisions to the IR, particularly in Section 5, will provide greater clarity in the definition of IR determinations and applicant responsibilities.	It is believed that the response to this NRC comment fully resolves the concerns raised by the NRC. NUMARC requests that the NRC consider this comment "closed" based upon the response.

PWR INTERNALS IR: RESPONSE TO NRC COMMENTS

NRC COMMENT #	NRC COMMENT	INDUSTRY RESPONSE	STATUS	PROPOSED IR REVISION	REMARKS
G-2	The discussion for many of the degradation mechanisms concludes that an applicant seeking license renewal need not evaluate these mechanisms further, beyond assurance by the license renewal applicant that no plant-specific design and operating features exist which would preclude the applicant from verifying these conclusions. It is not clear from these statements what the IR requires the applicant to do. How can one verify the conclusions without performing an evaluation? The IR should provide more specific instructions to the users and clearly define the criteria for screening plant specific features. Otherwise, the evaluations presented in it will have little value.	In accordance with agreements reached between NRC staff and NUMARC representatives on July 24, 1990, the common sentence referred to in the comment will be replaced by different language that provides more specific guidance to license renewal applicants regarding the review of plant-unique design features, construction records, and operating and maintenance history.	Closed	Section 4 of the report will be modified with the following text: "License renewal applicants intending to reference these generic conclusions are responsible for review of plant records, in order to assure that there are not deviations from the assumptions and criteria used in the report. This review should compare the design basis for particular components with the representative design bases given in Sections 3.1 and 3.2, including component material, fabrication and construction records. The component operating history should also be compared to the generic performance parameters described in Section 3.3. Finally, the specific criteria used in this section should be examined to assure that they, or equivalent criteria, apply to the component under consideration.	It is believed that the response to this NRC comment fully resolves the concerns raised by the NRC. NUMARC requests that the NRC consider this comment "closed" based upon the response. This comment is similar to comments received for other IRs, for example comment G-2 from the BWR containment IR. The same response is provided for which Agreement-in-Principle was reached from the February 20, 1991 meeting with the NRC.

PWR INTERNALS IR, RESPONSE TO NRC COMMENTS

NRC COMMENT #	NRC COMMENT	INDUSTRY RESPONSE	STATUS	PROPOSED IR REVISION	REMARKS
G-3	The IR, as written, does not clearly specify the assumptions that were made in performing the degradation analysis presented in it. Since the IR is expected to be widely used for license renewal purposes, the report should include a complete listing of all assumed parameters and, preferably, a user-oriented checklist such that the user can verify the applicability of the assumptions that were made in performing the degradation analyses.	We have agreed to provide clear criteria in all of the IRs that license renewal applicants can use to determine the applicability of the technical conclusions reached in Section 4 and 5. These criteria will take the form of if.....then statements, where possible, that spell out the requirements to be met. We believe that this IR contains the criteria needed, but we agree that the criteria will be reviewed for clarity of presentation. Specific cases of unclear criteria are addressed later in the specific comment responses.	Open	Changes to the report, as necessary, to clarify criteria, using NUMARC standard language.	It is believed that the response to this NRC comment fully resolves the concerns raised by the NRC. NUMARC requests that the NRC consider this comment "closed" based upon the response. This comment is similar to comments received for other IRs, for example G-11 from the BWR Containment IR. The same response is provided for which Agreement-in-Prinicple was reached from the February 20, 1991 meeting with the NRC.
G-4	A large number of components are dismissed in the IR from consideration for license renewal based on the arguments that these components do not perform any safety functions. While these components may not have direct safety functions to play in a normal operating condition or a design basis accident, their failure could jeopardize monitoring of the reactor or produce loose parts that could prevent safety related components from functioning properly. If failure of these components could prevent safety-related components from functioning properly, are the components important to license renewal? The IR should address this issue by either demonstrating these components are not important to license renewal or providing the technical basis for their dismissal.	The IR will be revised to eliminate the valuation of component safety significance now contained in Section 3. Instead, the report will identify clearly the internals components that are within its scope and those that are outside its scope. All components that are within the scope of the report will be described in Section 3.1, have its design bases described in Section 3.2, any relevant operating history described in Section 3.3, and be evaluated with respect to age-related degradation in Section 4. No determination will be made with respect to whether the components are important to license renewal. Such determinations are the responsibility of license renewal applicants.	Open	Substantial revisions will be necessary for Sections 2 and 3.	It is believed that the response to this NRC comment fully resolves the concerns raised by the NRC. NUMARC requests that the NRC consider this comment "closed" based upon the response.

PWR INTERNALS IR: RESPONSE TO NRC COMMENTS

NRC COMMENT #	NRC COMMENT	INDUSTRY RESPONSE	STATUS	PROPOSED IR REVISION	REMARKS
G-5	The terms wear and fretting [illegible] used somewhat loosely in the IR. Many of the components are stable in a reactor environment because of the presence of protective oxides. If these are removed by abrasion, rapid local oxidation (corrosion) can occur as the oxide films reform. This should be discussed in the IR, since it may play a role in the aging degradation phenomena.	The differentiation between "wear" and "fretting wear" as used in the IR is explained in Section 4.7.1., "wear" being a general term and "fretting wear" referring specifically to wear resulting from vibratory action. Both result from the relative motion of two components at their interface. Whether the material directly removed is the metal or its oxide is of secondary importance to the IR; the net effect is the same - a net loss of base material. However, Section 4.7.1. will be revised to reflect the fact that wear can result from either surface oxide removal or the direct removal of base material.	Open	Section 4.7.1. will be rewritten as follow: "Wear refers to the removal of material from either or two components due to mechanical interaction at their interface. The mechanical interaction can be sliding or impacting or both. In some cases, base material is directly removed by th[illegible] interaction; in other cases, a surface oxide layer is formed and removed during the wear cycle. The effect is the same: a net loss of base material. Wear which is the result of flow-induced vibration is referred to as fretting wear. Another type of wear which may occur in pressurized water reactors and is not related to flow-induced vibration is that associated with the intentional displacement of adjacent components, such as the stepping of control rods."	

NRC COMMENT #	NRC COMMENT	INDUSTRY RESPONSE	STATUS	PROPOSED IR REVISION	REMARKS
G-6	Many combinations of significant degradation mechanisms and components have been dismissed from consideration for license renewal due to the reason that the respective NSSS vendor has recommended periodic inspection programs. Such recommendations alone are not adequate justification for license renewal. As a minimum, the IR should include guidance to be used by a licensee to answer the following questions: (a) Whether and how were these recommendations implemented? (b) How effective were the recommendations to correct situations? (c) Has the overall safety margin of the plant been compromised as a result of operating the plant in the degraded condition prior to implementation of the vendor's recommendations? (d) Since the extent and frequency of the recommended inspection were apparently based on the current licensing term, how effective will the same inspection be through the license renewal period given that the plant will be further aged?	The degradation mechanisms and internals components currently closed out based on vendor recommended, plant/component specific programs are SCC of selected Alloy X-750 and Alloy A-286 components and stress relaxation of preloaded bolts. In those cases involving SCC of Alloy X-750 and Alloy A-286 component replacement and inspections have been recommended. In those plants which have replaced components, inspections show no additional SCC degradation has occurred. The affected components have not compromised the overall safety margins of the plants. For cases involving stress relaxation of preloaded bolts, the mechanical redundancy argument can be used on a plant specific basis (Section 6) to demonstrate that degradation will not cause loss of function of the system that compromise the overall safety margin of the plant.	Open	Section 5.3 of the IR will be modified to state, the following: A license renewal applicant is responsible for reviewing their related plant-specific documentation, in order to assure that the vendor plant specific elements required to mange the effects of SCC on Alloy X-750 and Alloy A-286 components are committed for use at their plant. Section 6.0 of the IR will be modified to include stress relaxation of preloaded bolts which cannot be inspected by current and effective methods. Aging management strategies including inspection or mechanical redundancy assessment will be recommended.	

NRC COMMENT #	NRC COMMENT	INDUSTRY RESPONSE	STATUS	PROPOSED IR REVISION	REMARKS
		Moreover, it is recognized that research is continuing to better characterize and inspect for relaxation of preloaded bolts. Quantitive inspections for stress relaxation are not currently in use; therefore, the IR will be modified such that stress relaxation of preloaded bolts is placed in Section 6. It should be recognized, however, that inspections to detect stress relaxation will in most cases be geared toward detection of the effects of stress relaxation (e.g. - fret marks, loosening, etc.) rather than detection of the mechanism itself. In a limited number of cases, sufficient records of initial preload, bolt length, etc. may be available that a direct assessment of stress relaxation is possible.			It is believed that the response to this NRC comment fully resolves the concerns raised by the NRC. NUMARC requests that the NRC consider this comment "closed" based upon the response.
G-7	The degradation of certain components (e.g., support columns, bolts and pins) was considered acceptable in the IR since these components represent mechanically redundant systems. For example, it is argued that failure of a few bolts is acceptable since that may not constitute failure of the bolt assembly. Such qualitative arguments are inadequate to make a case for license renewal since it is customary in all engineering designs, especially in the nuclear industry, to provide some amount of redundancy and margin. As a minimum, the following should be addressed: (a) Be specific about the number of components (e.g., bolts) that are being sacrificed.	Because of the variation in reactor internals configurations, a generic quantitative analysis to demonstrate the efficacy of mechanical redundancy arguments would be an extensive effort. Practical considerations preclude this type of extensive undertaking. Hence, mechanical redundancy will not be used in Section 4 or 5 of the IR for closing issues. Mechanical redundancy assessments will be listed in Section 6 of the IR as one potential plant-specific aging management strategy.	Open	Revise IR to delete credit in Sections 4 and 5 for mechanical redundancy considerations as a means of closing issues. However, it will be noted in Chapter 6 that analysis based on mechanical redundancy is an approach that license renewal applicants may wish to consider on a plant specific basis.	It is believed that the response to this NRC comment fully resolves the concerns raised by the NRC. NUMARC requests that the NRC consider this comment "closed" based upon the response.

NRC COMMENT #	NRC COMMENT	INDUSTRY RESPONSE	STATUS	PROPOSED IR REVISION	REMARKS
	(b) Provide calculations to support acceptance of failure of the redundant components.				
	(c) Quantify the safety margin that is being sacrificed and justify the adequacy of the remaining safety margin.				
	(d) It was stated in the IR that for some components a periodic inspection will be adequate for detection of a progressive failure of the system. Provide discussions as to how the frequency of inspection was determined to guarantee a timely detection.				
	(e) Since all of the subject components are being equally aged and many of them are probably equally stressed, what is the rationale for not considering a simultaneous failure of a large number of such components resulting in an overall catastrophic failure of the system? The following two additional factors should also be considered in the evaluation: 1. Failure of some components will result in redistribution of loads and, probably, higher stresses in the remaining components. 2. Failure of any component may cause impact resulting in instantaneous higher stresses in the remaining components.				

PWR INTERNALS IR: RESPONSE TO NRC COMMENTS

NRC COMMENT #	NRC COMMENT	INDUSTRY RESPONSE	STATUS	PROPOSED IR REVISION	REMARKS
	(f) Failure of certain components in a system should be evaluated on a case-by-case basis. Therefore, it is not clear how the IR proposes acceptance of such failures on a generic basis resulting in dismissal of certain degradation mechanisms and internal components from consideration for license renewal. If the current licensing basis cannot be maintained for these components, this should be clearly stated and guidance should be provided to licensees to evaluate this issue.				
G-8	The IR does not include a discussion on the seismic and environmental qualification of the internals. The IR should demonstrate continued qualification of the components considering the degradation that will occur during the license renewal period. Both the structural integrity and the functional operability of the degraded components should be demonstrated under the design basis events (e.g., SSE and LOCA).	Since the yield strength of the materials of construction increases with increasing "age" due to irradiation and since the structural integrity of the component under design basis events has been assured by meeting stress limits, there would be an improvement in margin with aging. However, some components were also designed based on deformation limits. Because of potential embrittlement due to irradiation and thermal aging, the acceptability of the expected deformations should be confirmed, particularly for plants that have not converted to "leak-before-break" design basis.	Open	The IR will be revised to recommend in Section 6 that license renewal applicants confirm the acceptability of functional deformation limits under design basis events.	It is believed that the response to this NRC comment fully resolves the concerns raised by the NRC. NUMARC requests that the NRC consider this comment "closed" based upon the response.

PWR INTERNALS IR: RESPONSE TO NRC COMMENTS

NRC COMMENT #	NRC COMMENT	INDUSTRY RESPONSE	STATUS	PROPOSED IR REVISION	REMARKS
G-9	Since many of the internal components exhibit cracking, the IR recommends the use of fracture mechanics evaluation as a tool for license renewal. However, since inspection of the inner components, supports and attachment welds are difficult due to access limitation, it is not clear how a fracture mechanics evaluation can be reliably performed in addressing the significant degradation mechanisms such as fatigue, neutron embrittlement and IGSCC. A fracture mechanics analysis can, at most predict the critical size and the growth rate of a given crack (detected or postulated), provided appropriate crack growth models are available. For this reason, inspection should be performed to ensure that the actual crack growth rate is bounded by the predicted model. In addition, inspection is required to ascertain that the flaw does not reach a critical size that has been established by the fracture mechanics evaluation. The IR should clearly indicate how the fracture mechanics evaluation will be utilized as a tool for license renewal for inaccessible components.	We cannot agree that many of the internals components exhibit cracking since this has not been the experience with PWR internals. We agree with the reviewers comments on the need to couple the use of fracture mechanics to inspections or conservative assumptions. Background is presented in the following paragraphs: When aging degradation is identified for specific reactor internal components, fracture mechanics can be used as a generic tool in evaluation of flaws observed by inspection or from hypothesized assumptions. Fracture mechanics is a methodology for assessing the potential for crack extension by fatigue and ductile tearing and for crack instability. For the stainless steel internals under service conditions, crack instability is expected to be characterized by ductile tearing not brittle fracture. Elastic plastic fracture mechanics will most likely be required for fracture assessments. Applications may be made both to actual situations where cracks have been observed and to situations where the postulation of cracks is prudent, such as in inaccessible components or portions thereof. It can then be used in conjunction with inspection.	Open	Because fracture mechanics analysis methods for PWR internals to date have been performed on a plant specific basis, Sections 5.1.1.1.2 and 5.1.2.3. will be revised to incorporate the background presented above in the Industry Response and then moved to Section 6. Components which cannot be inspected by the current and effective procedures identified in Section 5 will be listed. Provisions for inspecting such components must be made or component aging management would be addressed in Section 6.	It is believed that the response to this NRC comment fully resolves the concerns raised by the NRC. NUMARC requests that the NRC consider this comment "closed" based upon the response.

NRC COMMENT #	NRC COMMENT	INDUSTRY RESPONSE	STATUS	PROPOSED IR REVISION	REMARKS
		For reactor vessel internal structural components, both fracture toughness (J_{IC} and J-R response) and fatigue crack growth rates (da/dn vs delta K) are required to make meaningful not-overly conservative evaluations of actual or postulated cracks. Research of irradiation effects on stainless steels has been in progress for over two decades and is continuing. Thermal aging of cast austenitic stainless steels has been investigated for a comparable length of time. Fatigue crack growth does not appear to be significantly affected by irradiation and thermal aging (G-9A). An environmental factor yet to be specified for the fatigue crack growth law of Section XI of the ASME Code is expected to account for irradiation and thermal aging based on available data and is under development.			

PWR INTERNALS IR: RESPONSE TO NRC COMMENTS

NRC COMMENT #	NRC COMMENT	INDUSTRY RESPONSE	STATUS	PROPOSED IR REVISION	REMARKS
		It is reasonable to expect that fluence-toughness-at-temperature mechanistic models will be available at the time of need. Time-temperature-toughness behaviors of the cast material response to thermal aging should also be in hand. However, it should be kept in perspective that even a forty-year life extension is not expected to degrade the material thoughness significantly beyond that of the initial forty years of service. The exponential characteristics of thoughness degradation from irradiation of austenitic stainless steels (also that of the cast material for thermal aging) clearly indicate this. For example the fracture toughness for a fluence of $1X10^{21}$ or $2 X 10^{21}$ n/cm^2 usually falls within the scatter-band of the data.			

NRC COMMENT #	NRC COMMENT	INDUSTRY RESPONSE	STATUS	PROPOSED IR REVISION	REMARKS
		For cracks observed by inspection in internal components, it is reasonable to assume that either viable toughness and fatigue-crack growth models are available or will be available in a timely manner for license renewal considerations. For observed cracks, integrity evaluation should be rather straight forward. For inaccessible components, engineering judgment must be made in postulating potential flaws. Convincing engineering judgments or postulated flaws can be based on material and material fabrication history, stress analysis results, response for similar application and general field experience. Traditionally the quarter thickness flaw approach has found favor with the ASME Code Committee and with regulatory bodies since they often have representation on the Code Committee and also often are called on to approve such code application in the licensing and safety arena. For an inspectable component (the reactor vessel), an example can be found in Appendix G of Section III of the ASME code. In ASME Code Case N-481, a fracture mechanics integrity analysis based on a quarter thickness flaw actually replaces volumetric inservice inspection.			

PWR INTERNALS IR: RESPONSE TO NRC COMMENTS

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		As a provisional flaw size in an inaccessible component, a quarter thickness flaw(conceptionalized to a specific geometry) is recommended. With such a postulated flaw size, a crack growth and crack stability analysis may be performed for inaccessible components. An alternate approach for inaccessible components would be to establish a critical flaw size by elastic plastic fracture mechanics and then determine the flaw size for the beginning of life extension. Subsequently postulate growth by fatigue to critical size at the end of the life extension. This would form a basis for conservatively estimating a tolerable flaw size for judging the likelihood of such a flaw existing in inaccessible components. REFERENCE: (G9-A) L.A. James,: Fatigue Crack Propagation in Austen tic Stainless Steels," Atomic Energy Review, Vo.. 14, No. 1, pp 37-86, 1976.			
G-10	The IR did not address creep or the effect of high temperature as a degradation mechanism that can lead to failure of internal components. For example, the core plates (top and bottom) can undergo large deformation due to high temperature and excessive bending stresses due to "bowing" of the plates and reduction of the opening sizes. The IR should examine the effect of creep and high temperature as mechanisms for license renewal purposes.	The IR does address the phenomenon of creep degradation in Section 4.5. Creep is not a concern for the types of alloys used for the PWR internal components at temperatures below 1000°F. The comment regarding large deformations due to high temperature is confusing. There is no known occurrence of "bowing" of the core plates.	Open	None	It is believed that the response to this NRC comment fully resolves the concerns raised by the NRC. NUMARC requests that the NRC consider this comment "closed" based upon the response.

PWR INTERNALS IR: RESPONSE TO NRC COMMENTS

NRC COMMENT #	NRC COMMENT	INDUSTRY RESPONSE	STATUS	PROPOSED IR REVISION	REMARKS
G-11	A description of the weld materials and processes and materials for other types of attachments is not available in the report. The IR should identify the materials and the failure mechanisms of the attachment of the internal components, and demonstrate their acceptability for license renewal. A table listing the materials used for each component and its attachment is recommended.	The PWR internals IR is intended to address three vendors and their specific types of internals. A complete list of specific weld materials and processes is not within the scope of the IR. However, the IR will be modified to include a table which provides categories of materials and examples of each.	Open	A table will be added to Section 3.0 of the IR to address categories of materials, including welds, and examples of each.	It is believed that the response to this NRC comment fully resolves the concerns raised by the NRC. NUMARC requests that the NRC consider this comment "closed" based upon the response. This comment is similar to co. ment G-22.

PWR INTERNALS IR: RESPONSE TO NRC COMMENTS

NRC COMMENT #	NRC COMMENT	INDUSTRY RESPONSE	STATUS	PROPOSED IR REVISION	REMARKS
G-12	The effect of life extension or any existing safety assessment study should be considered for license renewal. This should be addressed in the IR.	The evaluation of existing safety assessments, as identified in 10CFR54.3(a), is for the purpose of identifying systems, structures, and components (SCC) important to license renewal. The approach of the Industry Reports (IR) is not to make this determination. The IR's assumes that all components which are in the scope of the IR, for evaluation of age-related degradation, are important to license renewal. However, it is not the function of the IR to determine whethar or not a particular SSC is important to license renewal. It is the responsibility of the individual applicant in accordance with the final rule to determine which SSC for their plant are important to license renewal. If an SSC which is important to license renewal and has been explicitly evaluated in an IR and that has been approved by the NRC, then the applicant may choose to use the IR to justify their individual plant assessment (IPA). It would be inappropriate for an IR to attempt to supersede the regulatory authority of the Licence Renewal Rule by trying to predetermine which SSCs will be important to license renewal.	Open	Incorporate the revised Standardized Language in the IR.	It is believed that the response to this NRC comment fully resolves the concerns raised by the NRC. NUMARC requests that the NRC consider this comment "closed" based upon the response.

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		It is recognized in the IRs that an applicant needs to consider plant specific operating history. A plant specific operating history review may include, but is not limited to, safety assessments when evaluating whether or not the age-related degradation conclusions presented in an IR are valid for their specific plant. The applicant specific requirement to examine their operating history is intended to identify any technical basis which may have a time dependence for the evaluation of age-related degradation. If there are any plant specific time dependencies which would invalidate any of the IR conclusions, then the applicant would need to address those specific age-related degradation mechanism/components combinations individually for the license renewal term. The IR requirement for an applicant to verify plant specific operating history is clarified in the Standardized Language.			

PWR INTERNALS IR: RESPONSE TO NRC COMMENTS

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G-13	The use of linear extrapolation of cumulative fatigue usage factors to screen components may be unconservative and could exclude components which could suffer fatigue damage. The IR should evaluate and address this issue further.	The terminology "linear extrapolation of cumulative fatigue usage factors: must be defined to be extrapolation of number of transient cycles, if necessary, and the resummation of fatigue usage. Cumulative usage factors can only be extrapolated linearly when one set of transients dominates the fatigue evaluation. While this is often the case, the linear extrapolation of cumulative usage factors is, in general, unconservative. It should be noted that the number of assumed cycles that actually occur.	Open	The use of the term "linear extrapolation of cumulative fatigue usage factors" will be defined more precisely where the terminology is used.	It is believed that the response to this NRC comment fully resolves the concerns raised by the NRC. NUMARC requests that the NRC consider this comment "closed" based upon the response.
G-14	The PWR internals listed in Table 3-3 (page 3-60) are evaluated as not safety significant because they do not perform any safety functions and are excluded from component life evaluation. See General Comment No. 4. While these components do not have an active safety function, they should be considered important to license renewal if the failure of these components as a result of age-related degradation has the potential of impacting the safety function of other safety significant components. Furthermore, the determination of component safety significance should also consider the impact of loose parts resulting from component failure on the safe operation of the reactor plant. There have been cases of partial failure of thermal shield supports - Maine Yankee and St. Lucie.	The PWR internals components listed in Table 3-3 (page 3-60)are considered to be out of the scope of the IR.		Revise report to clearly state that these components are not within the scope of the IR.	It is believed that the response to this NRC comment fully resolves the concerns raised by the NRC. NUMARC requests that the NRC consider this comment "closed" based upon the response.

PWR INTERNALS IR: RESPONSE TO NRC COMMENTS

NRC COMMENT #	NRC COMMENT	INDUSTRY RESPONSE	STATUS	PROPOSED IR REVISION	REMARKS
G-15	In Section 5 of component life evaluation, all safety significant internals components with no history of degradation were determined to be adequately managed in detecting and mitigating potentially significant age-related degradation. This determination in a large part relies upon the performance of normal inservice inspection (ISI) that cracks in the components resulting from age-related degradation will be detected. The ISI of these components based on the ASME Code, Section XI, requires only visual examination VT-3 on the accessible surfaces with a frequency of once every 10 years. The main objective of visual examination VT-3 does not appear to be adequate to find the cracks in these components. To ensure adequate inspection of components susceptible to age-related degradation, VT-1 or surface examination of the threaded areas should be performed prior to end during the license renewal period, and at a frequency such that the potential crack growth will not exceed the allowed critical flaw sizes. Furthermore, the report did not address the structural integrity of internals components which are not accessible for normal ISI during refueling outages. The life evaluation of those inaccessible components should also be addressed.	The IR will be revised to identify clearly in Section 5 the program elements that manage the effects of potentially significant age-related degradation on PWR internals components. In some cases these program elements will be based wholly or in part on inservice examination requirements of the ASME Code Section XI, Subsection IWB. In other cases the program elements may be based on analytical evaluation of the potential for small flaws to grow to critical size under the loading and environmental conditions to which the component is subjected. The revisions will provide acknowledgement that the ASME Code bodies are preparing a set of inservice examination requirements specific to internals components through a new Section XI subsection (subsection IWG). This acknowledgement will in no way imply that current subsection IWB inservice examination requirements that are applied to internals components (examination category B-N-3)are inadequate. It is the intent of the visual examination requirements of the ASME Code that "crack-like" indications detected during VT-3 examinations represent a relevant condition. While there are some differences in the characteristics of VT-1 versus	Open	The IR will be revised in Section 5 to include a description of the current inservice examination requirements contained in ASME Section XI, Subsection IWB, along with a justification of the VT-3 examination procedures as they apply to accessible surfaces of internals components. The justification will include a discussion of flaw tolerance procedures that can be used to demonstrate critical flaw characteristics and the relationship to VT-3 examination demonstration requirements. A discussion of the proposed subsection IWG will be included in Section 5 of the IR, with more detail in Section 6. Plant-specific options for dealing with the examination of component surfaces that are not accessible, and which cannot be made accessible, will be provided in Section 6.	It is believed that the response to this NRC comment fully resolves the concerns raised by the NRC. NUMARC requests that the NRC consider this comment "closed" based upon the response.

NRC COMMENT #	NRC COMMENT	INDUSTRY RESPONSE	STATUS	PROPOSED IR REVISION	REMARKS
		VT-3 examinations, such as demonstrated character recognition height and minimum distance from eye to object, both VT-1 and VT-3 as applied to the examination of accessible surfaces of internals components are capable of detection of surface flaws of significant length. When combined with analytical demonstration of tolerance to such flaws, the examination procedures are further justified. It should be noted that any reportable condition from VT-3 examination requires further characterization, using surface/volumetric examination or other means, in order to justify continued operation. Components that are inaccessible for inservice examination, and that cannot be rendered accessible by removal of the core or of other internals require plant-specific consideration, as described in Section 6 of the IR. Also, justifications for exemptions/reliefs from inservice examination requirements are a plant-specific consideration.			

PWR INTERNALS IR: RESPONSE TO NRC COMMENTS

NRC COMMENT #	NRC COMMENT	INDUSTRY RESPONSE	STATUS	PROPOSED IR REVISION	REMARKS
G-16	It is well known that it takes time to initiate stress corrosion cracking and the cracking may be accelerated with the presence of sensitized materials, high tensile residual stresses resulting from cold working and welding, crevices, stagnant flow and the history of coolant contamination. The good operating experiences of austenitic stainless steel components in the primary water environment so far may not guarantee similar performance in the next 40 years. The IR should propose augmented inservice inspection of these components when the conditions that would promote SCC as mentioned above are present.	We agree that it takes time to initiate stress corrosion cracking (SCC) and that cracking may be accelerated by certain conditions. However, the IR clearly addressed the need for three conditions to promote SCC. The primary reason SCC is not deemed an age-related degradation mechanism for life extension is due to the primary coolant chemistry. In fact, much of the data generated by GE provides additional support as to the critical nature of oxygen in water affecting the corrosion potential, thus SCC initiation on internal components. It is expected that 40 years of operation or even 20 years does provide a high level of confidence regarding the PWR primary coolant, the metal lurgical conditions of the internals and the various states of stress and the potential for SCC. Augmented inspections are not warranted.	Open	The IR will be modified to include more discussion regarding the PWR environment. Specifically, BWR experiments which have demonstrated the importance of low oxygen and low halogens will be added to section 4.0 of the IR to support that SCC is not a concern for license renewal.	It is believed that the response to this NRC comment fully resolves the concerns raised by the NRC. NUMARC requests that the NRC consider this comment "closed" based upon the response.

PWR INTERNALS IR: RESPONSE TO NRC COMMENTS

NRC COMMENT #	NRC COMMENT	INDUSTRY RESPONSE	STATUS	PROPOSED IR REVISION	REMARKS
G-17	Primary water stress corrosion cracking (PWSCC) of alloy 600 thermal sleeves were reported in several domestic PWR pressurizers. Therefore, component life evaluation for PWR internals should include components made of alloy 600.	No PWR internals components made of Alloy 600 have been identified in the report for any of the designs of the three domestic PWR vendors. Ni-Fe-Cr alloys are used for some components, notably for bolting and other mechanical fasteners. Stress corrosion cracking (SCC) of these Ni-Fe-Cr fasteners, as the result of a combination of material susceptibility, applied stress and reactor coolant environment, is described in some detail in Section 4.2 of the report. In particular, the 70Ni-16Cr-7Fe Alloy X-750 (SB-637, Grade N07750) is discussed on p. 4-9. SCC (or PWSCC) was deemed to be potentially significant for fasteners made of this material, with further evaluation described in Section 5.3. While the discussion does not identify PWSCC, the issue is SCC of PWR internals. We believe that this covers the topic.	Open	Should other Ni-base alloys be identified during the review and revision of the IR, the affected components will be described and the potential degradation evaluated in Sections 4.2 and 5.3. No revision to the IR is needed to cover the discussion of SCC (or PWSCC) for the Ni-Fe-Cr alloys identified to date.	It is believed that the response to this NRC comment fully resolves the concerns raised by the NRC. NUMARC requests that the NRC consider this comment "closed" based upon the response.

NRC COMMENT #	NRC COMMENT	INDUSTRY RESPONSE	STATUS	PROPOSED IR REVISION	REMARKS
G-18	The IR often does not give the specifics on how certain conclusions were reached, or demonstrate that the processes proposed to manage degradation are actually feasible. For example, the primary tools for the management of the issues identified in Section 5 are inspection and fracture mechanics analysis, but neither the report nor the references seem to contain the scoping calculations that are necessary to demonstrate that the cracks of concern can in fact be detected with reasonable reliability by the inspection techniques available.	Because of the variation in internals configurations and loadings, it is impractical to include the specifics on the inservice examinations and fracture mechanics evaluations that have been or that could be used to demonstrate that potentially significant age-related degradation is adequately managed. However, it is the procedure, rather than the results of application of the procedure, that is at issue in Section 5 of the IR. This procedure consists of the following steps: (1) inservice visual examination at periodic intervals, with a requirement that all relevant conditions be reported and evaluated; (2) elastic or elastic-plastic fracture mechanics calculations to show that "crack-like" relevant conditions do not threaten the integrity of the component; and (3) plant-specific action for those cases that cannot be shown to conform to (1) and (2). The criteria for Step (1) are already contained in the ASME Code Section XI, including remedial actions such as repair and replacement. Criteria for Step (2) are not contained explicitly in the ASME Code Section XI, but can be adapted from related Code Cases and non-mandatory appendices of Section XI.	Open	Section 5 of the IR will be revised to clearly delineate the steps in the programs used to manage the effects of potentially significant age-related degradation. For those cases where a combination of inservice examination and fracture mechanics can be used to manage degradation, the criteria for program element effectiveness will be described. This will include citing reference flaw sizes, flaw growth curves, and margins against flaw instability contained in the ASME Code Section XI, relevant Code Cases, and relevant non-mandatory appendices.	It is believed that the response to this NRC comment fully resolves the concerns raised by the NRC. NUMAHC requests that the NRC consider this comment "closed" based upon the response.

PWR INTERNALS IR: RESPONSE TO NRC COMMENTS

NRC COMMENT #	NRC COMMENT	INDUSTRY RESPONSE	'S	OPOSED IR REVISION	REMARKS
		Criteria for Step (3) must be justified by each applicant on a plant-specific basis. It is within the scope of the IR to describe the process in sufficient detail so that its effectiveness can be determined by the staff. It is beyond the scope of the IR to provide component-specific applications or results of application of the process steps.			
G-19	In general, the IR asserts that stress corrosion cracking of stainless steels is not a significant age-related degradation mechanism in PWRs (except for a few specific cases where cracking has occurred in service) because of the relatively benign environment. However, crevices are known to promote stress corrosion type cracking (even in the absence of high stress) in stainless steels and other austenitic alloys even when the bulk environment is of relatively high purity. The report should include discussions of components with crevices or creviced geometries and evaluate the potential for crevice corrosion cracking.	It is agreed that so e discussion regarding creviced type geometries within the PWR internals is needed. Much of the available literature regarding the potential for crevices concentrating in a local environment is for oxygenated and/or oxidizing environments. There is no know data to suggest that PWR primary coolant type chemistries, e.g. low oxygen and low halogens, will promote SCC on creviced type geometries.	Open	The IR will be modified to include a discussion on crevice corrosion. Unless data to the contrary is found, the potential for crevice corrosion degradation will be considered managed if proper primary coolant chemistry is maintained.	It is believed that the response to this NRC comment fully resolves the concerns raised by the NRC. NUMARC requests that the NRC consider this comment "closed" based upon the response.

NRC COMMENT #	NRC COMMENT	INDUSTRY RESPONSE	STATUS	PROPOSED IR REVISION	REMARKS
G-20	The mechanical redundant feature is used in the IR to demonstrate acceptable performance of internal components having multiple supports during the license renewal term. However, it should be noted that age-related degradations are common mode failure mechanisms, and they could cause simultaneous failures in a multiple-support system. The IR lacks justification to support the assertion that PWR internals are not susceptible to common mode age-related failure mechanisms. The IR should be revised to address this issue. See General Comment No. 7.	The concern about common mode failure mechanisms is similar to Part (e) of Comment G-7. The variability within a bolted joint is considerable, with the examples of bolt hardness and bolt preload being typical. It is not unusual to see bolt hardness (a surrogate for yield strength) vary by 40% in a bolted joint, corresponding to a yield stress variation from approximately 100 to 130 ksi. Similarly, depending on the torquing procedure, bolt preload can vary by ±30% (four passes) to a factor of 2 (three passes). With such variability, only the application of an extreme (Level D) load would challenge a substantial number of bolts simultaneously. For nominal operating conditions, preferential degradation is expected.	Open	None	It is believed that the response to this NRC comment fully resolves the concerns raised by the NRC. NUMARC requests that the NRC consider this comment "closed" based upon the response.
G-21	Other than defining the interface boundary between internals and the pressure vessel, the report does not address age-related degradations at the pressure boundary regions. The IR should address potential problems at these regions.	The interfaces between the reactor internals and the pressure boundary (reactor vessel) are all readily inspectable, if needed. The reactor vessel internals do not include the pressure boundary. However, because they are not covered in other IRs the BMI thimble tubes have been included in the Internals IR and are discussed in Sections 3.1.2.5 and 5.5. In addition, interface boundary degradation at the radial keys, upper core plate alignment pins and internals/vessel flanges are discussed.		None	It is believed that the response to this NRC comment fully resolves the concerns raised by the NRC. NUMARC requests that the NRC consider this comment "closed" based upon the response.

PWR INTERNALS IR: RESPONSE TO NRC COMMENTS

NRC COMMENT #	NRC COMMENT	INDUSTRY RESPONSE	STATUS	PROPOSED IR REVISION	REMARKS
G-22	Materials for major PWR internals are specified as austenitic stainless steel or nickel-based alloys. This includes a large group of materials and their susceptibility to age-related degradations may vary. Effects of aging degradations on core internals cannot be evaluated without knowing the specific materials of construction. The IR should be revised to clearly identify the material and the potential age-related degradation mechanisms for each major PWR internals.	It is understood that the specific materials of construction, for each component for the three vendors various designs would be too cumbersome for this IR. It is also noted that in considering material degradation, material categories should be sufficient. A specific category will cover materials of similar chemical composition. More detail, in the form of material categories for the internals will be added.	Open	A table will be added to Section 3.0 of the IR to address material categories used in the construction of PWR internals. Each category will be briefly discussed in the text of the IR and examples of each category will be provided, e.g. Austentic Stainless Steels would be a category for materials such as Types 304 and 316.	It is believed that the response to this NRC comment fully resolves the concerns raised by the NRC. NUMARC requests that the NRC consider this comment "closed" based upon the response. This comment is similar to comment G-11.

PWR INTERNALS IR: RESPONSE TO NRC COMMENTS

NRC COMMENT #	NRC COMMENT	INDUSTRY RESPONSE	STATUS	PROPOSED IR REVISION	REMARKS
S-1	Both in-reactor tests and field experience with stainless steel fuel cladding have shown that irradiation-assisted-stress-corrosion cracking (IASSC) can occur in PWRs. This may indicate either that the local environments can be sufficiently aggressive to produce SCC problems or that the cracking does not require an aggressive, aqueous environment. Impurity segregation grain boundaries in some low-purity materials (in particular Type 304 SS) may be so extensive at high fluence that the materials will crack even in inert environments if the stress level is sufficiently high. Until the mechanism of IASCC is better understood and it can be shown that the electro-chemical potential is low throughout the core and upper internal region, all stainless steels with fast fluence $5x10^{20}$ n/m^2 should be considered potentially susceptible to IASCC.	We agree that irradiation-assisted stress corrosion cracking (IASCC) occurs in stainless steel cladding of PWR fuel elements. Detailed analysis of these situations has shown that local stress states caused by fuel relocation or pellet-clad interaction were the significant factor. An adequately severe combination of susceptible material (i.e., irradiated beyond the threshold of $5x10^{20}$n/cm^2), high stresses and an aggressive environment is needed. For components irradiated to levels below $5x10^{20}$n/cm^2 and exposed to PWR coolant conditions, the stresses must be quite large (e.g., greater than 10 ksi) or the component must contain flaws of some size (i.e., greater than 2 inches in length). BWR internals components have been evaluated using degraded material properties caused by exposure to fluences of $5x10^{20}$n/cm^2 (ductility and elastic-plastic fracture toughness) and with design stresses. Postulated flaws do not lead to applied stress intensities sufficient to cause crack initiation. Based on these screening calculations, the criterion that: 1. exposure is less than $5x10^{20}$n/cm^2; 2 the environment is controlled to PWR operating limits; and 3. the operating stresses are within the design envelope.	Open	Section 4.3 will be revised to clarify the threshold levels and criteria used to support the finding of non-significance.	It is believed that the response to this NRC comment fully resolves the concerns raised by the NRC. NUMARC requests that the NRC consider this comment "closed" based upon the response.

PWR INTERNALS IR: RESPONSE TO NRC COMMENTS

NRC COMMENT #	NRC COMMENT	INDUSTRY RESPONSE	STATUS	PROPOSED IR REVISION	REMARKS
		Leads to a finding of non-significance. For stress levels that are moderate to high, the threshold for significant exposure is $1x10^{20} n/cm^2$.			
S-2	Page 1-21, top paragraph: The IR states that another set of component-degradation mechanism combinations is considered to be generically resolved because the potentially significant age-related degradations can be shown to be bounded within established limits by currently acceptable maintenance, inspection, test, and analytical evaluation procedures which are in place at the various utilities. How could the potentially significant mechanisms be resolved generically without plant-specific assessment of the degradation levels and management programs? The IR should either provide a detailed generic discussion or keep these issues open for resolution on a plant specific basis.	The industry reports separate generic conclusions from any plant-specific actions by using a standardized format (e.g. Section 4 evaluates the potential significance of age-related degradation mechanism), standardized language that outlines license renewal applicant responsibilities, and definitive criteria for drawing generic conclusions. The intent of Section 4 for example, is to provide criteria that, if met by the applicant, represent a generic basis for the determination that a particular age-related degradation mechanism is insignificant for a particular component. The intent of Section 5 is to provide an adequate description of program elements that, if in effective use by the applicant, represent a generic basis for management of potentially significant degradation. Failure to satisfy the criteria in Section 4, or to have the program elements of Section 5 in place, requires the applicant to perform a plant-specific evaluation.	Open	The IR will be modified to incorporate the standardized language providing an outline of license renewal applicant responsibilities relative to report findings.	It is believed that the response to this NRC comment fully resolves the concerns raised by the NRC. NUMARC requests that the NRC consider this comment "closed" based upon the response.

PWR INTERNALS IR: RESPONSE TO NRC COMMENTS

NRC COMMENT #	NRC COMMENT	INDUSTRY RESPONSE	STATUS	PROPOSED IR REVISION	REMARKS
S-3	Page 1-25, top paragraph: Regarding the support columns, the IR states that the components are subject to limited crack growth, because of the presence of compressive stress apparently during normal operation. It is not clear whether an assessment has been made for possible existence of tensile stresses during other plant conditions. This should be verified and confirmed in the IR.	It appears the NRC concern is regarding the variation in the loading conditions of the support columns. While it is generally true that these components experience compressive stresses over most of their life, there may be short periods of time where some components will experience tensile stresses due to thermal transients or design basis events. A detailed study of this issue on a generic basis is outside the scope of the IR. However, crack growth of an existing flaw would be limited due to the short duration of the accident stresses and hence is inconsequential. The discussion on cracking and its growth will be removed from the top of page I-25.	Open	Revise the IR to indicate a need for license renewal applicants to confirm the absence of tensile stresses large enough to result in significant crack growth in support columns; also, revise the top of page I-25 to eliminate the discussion of cracking and its growth.	It is believed that the response to this NRC comment fully resolves the concerns raised by the NRC. NUMARC requests that the NRC consider this comment "closed" based upon the response.

NRC COMMENT #	NRC COMMENT	INDUSTRY RESPONSE	STATUS	PROPOSED IR REVISION	REMARKS
S-4	Page 1-26, second and third paragraph: Regarding fatigue evaluation, the IR states that re-analysis and an acceptable in-service inspection program would be carried out, and as a result it concludes that fatigue is not a significant age-related degradation mechanism. A degradation mechanism cannot be eliminated generically at present based on plant-specific results of future re-analysis and inspection programs. The IR should be based on results which are known at present. In addition, if the results are plant-specific, the IR should describe plant-specific evaluation and should not dismiss the degradation mechanism generically. Fatigue is an open issue.	Section 5 of the IR describes the elements of a fatigue analysis/reanalysis and inservice inspection program for fatigue-critical internals that, if shown by a license renewal applicant to be in effective use, provides a generic basis for management of potentially significant fatigue damage. The IR is not required to report the results of an established program application; however, the IR <u>does</u> report the results of the application of the B&W fatigue reanalysis (see IR References 4-3, 4-37, and 4-38) in order to demonstrate the effective nature of the process. Such a demonstration gives a measure of confidence to license renewal applicants that the use of fatigue reanalysis will result in low fatigue usage factors for the license renewal term.	Open	None	It is believed that the response to this NRC comment fully resolves the concerns raised by the NRC. NUMARC requests that the NRC consider this comment "closed" based upon the response.
S-5	Page 3-5 and 3-45: The IR lists a few reactor components that are not included in the subject IR. This IR should provide references to other pertinent IRs that address the excluded components. This will ensure that the excluded components have not been missed.	Components that are not included are out of the scope of the IR; cross-references other IRs will not be provided.	Open	None	It is believed that the response to this NRC comment fully resolves the concerns raised by the NRC. NUMARC requests that the NRC consider this comment "closed" based upon the response.

PWR INTERNALS IR: RESPONSE TO NRC COMMENTS

NRC COMMENT #	NRC COMMENT	INDUSTRY RESPONSE	STATUS	PROPOSED IR REVISION	REMARKS
S-6	Page 3-27, fourth paragraph: The IR generically eliminated the need for evaluation of the Westinghouse secondary core support assembly (and similarly the Combustion Engineering core stop interface) based on the assertion that its function is to provide core support in the unlikely event of core barrel failure. It is not justifiable to accept failure of such an important component, though redundant, and lose the existing safety margin provided with the original design. The degradation of this component should be evaluated for license renewal. See General Comment No 7.	The IR will be revised to indicate that the secondary core support is a component which is out-of-scope of the IR.	Open	Section 2 of the IR will be revised to indicate that the secondary core support is a component which is out-of-scope of the IR.	It is believed that the response to this NRC comment fully resolves the concerns raised by the NRC. NUMARC requests that the NRC consider this comment "closed" based upon the response.

PWR INTERNALS IR: RESPONSE TO NRC COMMENTS

NRC COMMENT #	NRC COMMENT	INDUSTRY RESPONSE	STATUS	PROPOSED IR REVISION	REMARKS
S-7	Page 3-39, last paragraph: Regarding the hold-down spring the IR concludes that the component is not safety-significant, and no further evaluation is required. However, the hold-down spring is needed to provide vertical core restraint. Moreover, degradation of the hold-down spring has been observed in past inspections resulting in wearing of the reactor vessel flange and head (Page 3-54). Therefore, it does not seem appropriate to classify the hold-down spring as non-safety-significant, and not to evaluate the degradation effects. The IR should address this issue.	The IR used the term "not safety significant" to imply that component failure would not adversely affect the ability to shut down the plant or to maintain a coolable geometry. The holddown springs meets this test. The only postulated degradation mechanisms for the holddown spring is relaxation. There currently exists methods and standards for monitoring of internals for holddown spring relaxation. In addition, visual inspections and preload measurements can be readily performed, if a need is determined to exist at a particular plant. The rationale for eliminating holddown springs as safety insignificant is provided on page 3-39, where it is also noted that other components are available to take any loads a degraded holddown spring does not. However, the IR will be revised to not make a determination on whether holddown springs are safety significant or important to license renewal. Holddown springs will be outside the scope of the IR.	Open	The IR will be revised so that holddown springs are outside the scope of the IR.	It is believed that the response to this NRC comment fully resolves the concerns raised by the NRC. NUMARC requests that the NRC consider this comment "closed" based upon the response.

PWR INTERNALS IR: RESPONSE TO NRC COMMENTS

NRC COMMENT #	NRC COMMENT	INDUSTRY RESPONSE	STATUS	PROPOSED IR REVISION	REMARKS
S-8	Page 3-40 to 3-44, Sections 3.1.2.4 to 3.1.2.7: Thermal shields, incore instrumentation guide tubes, irradiation specimens guides and inlet flow distribution components are all included as elements in the three generic regions for core internals as given in Tables 3-2 (a-c). In order to have a consistent report format and to avoid unnecessary confusions, these components should be included as subsections to the three generic core regions.	We agree with comment. However, the report will no longer assess whether these components are important to safety. They will either be in or out of the scope of the IR. Note that thermal shields, the non-pressure boundary portions of the instrument guide tubes, the irradiation specimens guides, and inlet flow distribution components will be considered outside the scope of the IR.	Open	Modify section 2 of the IR to remove thermal shields, nonpressure boundary portions of the instrument guide tubes, the irradiation specimen guides, and the inlet flow distribution components from the scope of the IR.	It is believed that the response to this NRC comment fully resolves the concerns raised by the NRC. NUMARC requests that the NRC consider this comment "closed" based upon the response.
S-9	If the age-related degradation could result in the loss of the surveillance samples, the guides must be considered for component life evaluation. Page 3-43, Section 3.1.2.6 Irradiation Specimen Guides: The IR classifies the irradiation specimen guides and holders as non-safety-significant. Failure of these components may result in mismanagement of the irradiation specimens which are required by regulations and are fundamental to monitoring the fluence level of the reactor pressure vessel (RPV) and other components. A reliable irradiation estimate of the RPV is needed for its life evaluation. The IR should evaluate and address these components.	The IR will be revised to indicate that irradiation specimen guides are out-of-scope of the IR.	Open	The IR will be revised to indicate that irradiation specimen guides are out-of-scope of the IR.	It is believed that the response to this NRC comment fully resolves the concerns raised by the NRC. NUMARC requests that the NRC consider this comment "closed" based upon the response.

NRC COMMENT #	NRC COMMENT	INDUSTRY RESPONSE	STATUS	PROPOSED IR REVISION	REMARKS
S-10	Page 3-45, first paragraph, last sentence: The IR states that although the majority of PWR internals were fabricated prior to implementation of Subsection ND, the design philosophies of the three PWR vendors were such that the intent of Subsection NG was met. Certain requirements of Subsection NG might not have been met. The IR should include a discussion identifying the NG requirements which were not satisfied by the earlier designs and evaluate its impact on license renewal.	The reviewers' concern about potentially not having met all NG requirements applies equally to the current licensing basis as well as to the license renewal period. Since the current licensing basis has been deemed adequate, it is not obvious that further effort is required to address this comment. However, since no credit was taken for this statement, the IR will be revised to delete this statement in order to minimize future confusion.	Open	Delete the indicated statement on page 3-45.	It is believed that the response to this NRC comment fully resolves the concerns raised by the NRC. NUMARC requests that the NRC consider this comment "closed" based upon the response.
S-11	Pages 3-46 and 3-47, items a,b,c: Regarding the earthquake forces, the IR uses the terms design and hypothetical apparently to indicate operating and safe shutdown, respectively. If so, the IR should be revised by using the currently acceptable terminology.	We agree with the comment and will revise IR to use standard terminology (OBE and SSE).	Open	Revise IR to use standard terminology (OBE and SSE) on pages 3-46 and 3.47.	It is believed that the response to this NRC comment fully resolves the concerns raised by the NRC. NUMARC requests that the NRC consider this comment "closed" based upon the response.
S-12	Section 3.2.3.1 Leak-Before-Break: For plants approved for leak-before-break with piping fabricated from cast stainless steel, the effect of thermal aging on cast stainless steel should be evaluated to ensure that the leak-before-break acceptance criteria would be satisfied throughout the license renewal term.	We believe that this comment is relevant to the reactor coolant systems IR and not the internals. Leak-Before-Break (LBB) is presented as an example of analysis which, if performed on a plant's piping, can result in reduced loads on the internal components. It is agreed that if a plant has LBB for its current license, then that evaluation may need to be resubmitted for the license extension period.	Open	None	It is believed that the response to this NRC comment fully resolves the concerns raised by the NRC. NUMARC requests that the NRC consider this comment "closed" based upon the response.

NRC COMMENT #	NRC COMMENT	INDUSTRY RESPONSE	STATUS	PROPOSED IR REVISION	REMARKS
S-13	Section 4.1 Irradiation Embrittlement: This section indicates that irradiation embrittlement reduces that ductility properties, but it is not considered as a significant age-related degradation mechanism. The IR has not provided a fracture mechanics analysis to demonstrate that the PWR internals have adequate toughness after they have had their ductility reduced. The IR should provide such a demonstration by using the fracture toughness properties for weld metal irradiated to the end of license renewal period. A critical flaw size should be determined and IR should recommend an inspection method for detection of flaws and an inspection period to ensure flaws do not grow to a critical size. See General Comment No. 15.	A generic fracture mechanics analysis is outside the scope of the IR. However, the use of fracture mechanics, coupled with inspection, is a viable aging management tool.	Open	Remove discussion on top of page 4-6 and change Section 6 to address fracture mechanics and inspections on a plant specific basis.	It is believed that the response to this NRC comment fully resolves the concerns raised by the NRC. NUMARC requests that the NRC consider this comment "closed" based upon the response. Also, see responses to comments S-14e, G-9, and G-15.

PWR INTERNALS IR: RESPONSE TO NRC COMMENTS

NRC COMMENT #	NRC COMMENT	INDUSTRY RESPONSE	STATUS	PROPOSED IR REVISION	REMARKS
S-14	Page 4-4 through 4-7:				
	(a) The IR presents experimental data to show how material properties change after prolonged exposure to neutrons. The data indicate that for the irradiated material, the yield strength approaches the ultimate strength and the uniform elongation and the total elongation drop substantially from their unirradiated values. However, the IR concludes that there still remains a significant amount of ductility. It appears that the data do not support the conclusions. In addition, the most important parameter of concern for evaluation of cracked components is the fracture toughness which is substantially reduced due to neutron embrittlement. The IR should provide more details about the experiments (e.g., fluence, flux and temperature for BWR; fracture toughness and tearing modulus for PWR), and reevaluate by considering the appropriate materials and the end-of-life fluence level at a realistic flux level.	This IR, and a related IR covering BWR internals components, present experimental data on mechanical properties of irradiated austenitic stainless steel, showing that the yield strength increase and the elongation decreases substantially. However, the issue is not the magnitude of change, but whether the residual mechanical properties are adequate to assure structural integrity of the component during the license renewal term. Fracture toughness data for irradiated stainless steel also indicates substantial reduction, from very high elastic-plastic J_{IC} values of 2000 to 5000 in-lb/in^2 and very steep J-R curves. With irradiation to high fluences (>10^{21} n/cm^2, E > 0.1 Mev), J_{IC} is reduced to values on the order of 100 to 200 in-lb/in^2. The tearing resistance is also degraded. These data have been obtained under conditions representative of BWR and PWR service conditions, and indicate a saturation effect at the higher fluence levels that mitigates concerns about end-of-life fluence. More data will be accumulated on mechanical properties of irradiated stainless steels in the future that will improve our understanding of	Open	Section 4.1.1 will be revised to include a discussion of mechanical property changes as a result of neutron exposure, including changes in fracture toughness. The section will also be revised to include a qualitative discussion of weld metal chemistry, and its effect on embrittlement, and of the potential for damage to non-stainless steel bolting, due to its location within the reactor vessel. (This proposed IR revision applies to parts a,b, and c of this comment).	

NRC COMMENT #	NRC COMMENT	INDUSTRY RESPONSE	STATUS	PROPOSED IR REVISION	REMARKS
		accelerated aging effects. However, the present data base is adequate for the purpose of determining those components that are subject to potentially significant neutron irradiation embrittlement.			
	(b) The materials in the heat affected zone (HAZ) usually suffer severe neutron embrittlement. The IR did not address this issue. The IR should consider the irradiation embrittlement effect on the HAZ materials.	The materials that comprise PWR internals are, with the exception of some bolting, austenitic stainless steel and Ni-Fe-Cr (nickel-base) alloys. Weld metals for these materials do not contain the trace constituents that would enhance irradiation embrittlement over and above that observed for base metal.			
	(c) Certain parts of the internal components (e.g., bolts, welds, other attachments) are probably not made of stainless steel. The IR does not address neutron embrittlement of these components and should do so.	Non-stainless steel bolting is not used in the high flux regions of the core. The IR will be revised to include a discussion of the location of internals bolting relative to high flux regions.			
	(d) For fracture mechanics analyses, the IR refers to Sections 5.1.1.3 and 5.1.4.1.1. However, the IR does not contain these sections. Section 5.1.1.1.2 which deals with analysis of flaws should contain quantitative description of the influence of PWR internal environment on fracture toughness and crack growth rates for life extension purposes.	The reviewer has correctly identified typographical errors. The reference to Sections 5.1.1.3 and 5.1.4.1.1 should be to Sections 5.1.2.3 and 5.1.1.1.2. As noted in the response to S-14(a), the IR will be revised to include more extensive discussion of irradiation-induced mechanical property changes in Section 4. The use of these data for aging management evaluations is covered under the response to S-14(e).		Revise page 4-6 to properly reference Sections 5.1.2.3 and 5.1.1.1.2. Revise Sections 4, 5 and 6 to clearly explain the criteria for insignificance, the aging management process steps, and any plant-specific options that are at a stage of development to warrant inclusion in Section 6.(This proposed IR revision applies to parts d,e, and f of this comment).	

NRC COMMENT #	NRC COMMENT	INDUSTRY RESPONSE	STATUS	PROPOSED IR REVISION	REMARKS
	(e) The IR states that irradiation embrittlement of PWR internals materials in itself is not a significant aging degradation mechanism, since fracture mechanics analyses can be used for assessment. Availability of an assessment method alone is not an adequate ground for elimination of a degradation mechanism. The IR should provide a fracture mechanics analysis (either generic, i.e., enveloping all plant conditions or specifying that licensees referencing the IR in a renewal application must perform a plant-specific analysis) and a subsequent evaluation based on the analysis results.	For those internals components subject to potentially significant neutron irradiation embrittlement, Section 5 of the IR provides a description of program elements aimed at managing this form of degradation. These program elements consist of three steps: (1) inservice visual examination, in accordance with the requirements of the ASME Code Section XI, Subsection IWB; (2) more extensive volumetric or surface examination, in accordance with Code provisions, in order to characterize reported relevant conditions and their disposition; and (3) repair and/or replacement of those components subject to damage beyond that for which continued operation can be justified. The evaluation of "crack-like" relevant conditions consists of flaw growth calculations and a determination of continued structural integrity on a plant-specific bases. It is beyond the scope of the IR to provide generic or bounding results for such flaw evaluations for the wide variety of internal component designs, loadings and operating conditions. It is appropriate in the IR, however, to describe the evaluation process.	Open		

NRC COMMENT #	NRC COMMENT	INDUSTRY RESPONSE	STATUS	PROPOSED IR REVISION	REMARKS
		The periodic visual examination part of the process (VT-3) is described sufficiently in the Code to require no further discussion here. Supplementary visual, surfaces and volumetric examination procedures to characterize relevant conditions (e.g., a crack-like indication) are also described in the Code, with the utility required to justify the particular procedures employed to jurisdictional authorities. It would be inappropriate for the IR to prioritize particular supplementary examination steps. In lieu of supplementary examination, a utility might choose to conservatively characterize the relevant condition and demonstrate directly, without further examination, that the safety function of the component is not compromised. Such a plant-specific action is permissible in the Code and should not be precluded by the IR. Similarly, in lieu of supplementary examination, a utility might choose to repair or replace the affected component. Again, such a plant-specific action is permissible in the Code and should not be precluded by the IR.			

NRC COMMENT #	NRC COMMENT	INDUSTRY RESPONSE	STATUS	PROPOSED IR REVISION	REMARKS
		The IR does provide guidance with respects to evaluation process steps, and these steps will be stated clearly in Section 5. For example, while the justification for fatigue crack growth rates and fracture toughness of irradiated material is the responsibility of the license renewal applicant, it is expected that irradiation will have little effect on fatigue crack growth rates in austenitic stainless steels. Therefore, relatively modest adjustments to the fatigue crack growth rate curve in Appendix C of Section XI will be required. More important than the effect of irradiation is the effect of the water environment, which can be accommodated by introducing an environmental factor consistent with in-progress work in support of Appendix C. The effect of irradiation on fracture toughness is substantial, but the importance is mitigated by low stress levels, compressive loadings, and small flaw sizes. Each plant-specific application of the flaw evaluation process will have to provide justification of the fracture toughness used.			

PWR INTERNALS IR: RESPONSE TO NRC COMMENTS

NRC COMMENT #	NRC COMMENT	INDUSTRY RESPONSE	STATUS	PROPOSED IR REVISION	REMARKS
	(f) The IR concludes that in the absence of flaws, irradiation embrittlement is not a significant age-related degradation mechanism for the license renewal period, and no further evaluation is required. Since a flaw can be initiated due to other degradation mechanisms, the synergistic effect should be considered and the irradiation effect should not be dismissed from consideration for license renewal.	(f) Synergistic effects are included in the evaluation process, to the extent that service experience and laboratory data can justify. The visual examination is directed toward relevant conditions from any of a number of age-related degradation mechanisms, as are supplementary examinations. The flaw evaluation is directed toward a combination of cyclic crack growth, possibly accelerated by environmental effects, and flaw stability, which can be affected by other age-related degradation processes. We believe that synergistic effects have been included in the evaluation process, to the extent necessary. Irradiation effects are not dismissed from further consideration in the IR. Section 4 prescribes criteria that must be met in order that neutron irradiation embrittlement of internals components is not significant. For those components that fail to meet these criteria, neutron irradiation embrittlement is potentially significant and the aging management programs in			

PWR INTERNALS IR: RESPONSE TO NRC COMMENTS

NRC COMMENT #	NRC COMMENT	INDUSTRY RESPONSE	STATUS	PROPOSED IR REVISION	REMARKS
		Section 5 of the IR can be used for evaluation. For those license renewal applicants unable to demonstrate that the aging management programs in Section 5 are in place, plant-specific action are necessary. Some options in this regard are described in Section 6.			
S-15	Page 4-6, Section 4.2.1: The IR correctly points out that the oxygen level during operation and the halogen levels are low for occurrence of stress corrosion cracking (SCC). However, the sulfate from demineralizer resins and the oxygen level prior to and during shutdown are potential contributors to SCC. The IR should evaluate these two sources as well.	The initial part of the comment regarding oxygen levels prior to start-up is correct and the IR will include a discussion on this. The second part of the comment regarding sulfate excursions from the demineralizers is not a common plant condition. This can occur as a result of a plant specific situation. It is noted that changes in primary coolant pH and conductivity, beyond a plants technical specification limits will prompt evaluations to establish the source of the changes and corrections will be made.	Open	Section 4.2 of the IR will be modified to include a discussion on the oxygenated condition s which exist prior to start-up. The IR will reflect that halogen concentration must be maintained below 150 ppb at all times regardless of system temperature. Although oxygenated conditions may exist prior to start-up, halogens are kept below 150 ppb and the temperature can not exceed 180°F. A cold, oxygenated water with halogens below 150 ppb is not an environment which will promote SCC on PWR internal component materials.	It is believed that the response to this NRC comment fully resolves the concerns raised by the NRC. NUMARC requests that the NRC consider this comment "closed" based upon the response.

PWR INTERNALS IR: RESPONSE TO NRC COMMENTS

NRC COMMENT #	NRC COMMENT	INDUSTRY RESPONSE	STATUS	PROPOSED IR REVISION	REMARKS
S-16	Pages 4-7 to 4-9, Section 4.2.2: It is stated that stagnant conditions inside the pressure vessel could create a local environment which differs from the bulk PWR environment. There is no assurance that components made of type 304 stainless steel are not exposed to such a locally corrosive environment. The IR should be revised to address this potential degradation environment.	The sentence the reviewer is referring to is not applicable to internals and will be deleted from the IR. Crevice type corrosion will be discussed. Much of the available literature regarding the potential for crevices concentrating in a local environment, causing SCC on stainless steels, is for oxygen and/or oxidizing environments. There is not any know data to suggest that PWR primary coolant type chemistries, e.g. low oxygen and low halogens, will promote SCC on creviced type geometries within the PWR internals.	Open	A discussion of crevice corrosion will be added to section 4.2. of the IR. The sentence the reviewer is referring to will be deleted.	It is believed that the response to this NRC comment fully resolves the concerns raised by the NRC. NUMARC requests that the NRC consider this comment "closed" based upon the response. See also the response to comment G-19.
S-17	Page 4-9: Radiation hardening can make a material more susceptible to stress corrosion cracking. The IR should address this potential degradation environment.	Reviewer is referring to IASCC degradation which is addressed in Section 4.3 of the IR. We believe this is adequately addressed in the IR.		None	It is believed that the response to this NRC comment fully resolves the concerns raised by the NRC. NUMARC requests that the NRC consider this comment "closed" based upon the response.

PWR INTERNALS IR: RESPONSE TO NRC COMMENTS

NRC COMMENT #	NRC COMMENT	INDUSTRY RESPONSE	STATUS	PROPOSED IR REVISION	REMARKS
S-18	Page 4-10: Components fabricated from alloy 600 (e.g., flux thimble tubes) are potentially susceptible to stress corrosion cracking in the PWR coolant environment. The IR should address this potential degradation mode.	Westinghouse flux thimble tubes are made of austenitic stainless steel, not Ni-Fe-Cr materials such as Alloy 600. It is believed that Combustion Engineering and Babcock & Wilcox flux thimble tubes are also made of austenitic stainless steel. No Alloy 600 PWR internals components were identified during the preparation of this report. This finding will be reviewed in light of the staff comment. If such components are identified, Section 4 of the IR will be modified to include a discussion of PWSCC and its potential significance to PWR internals.	Open	Should PWR internals components made of Alloy 600 be identified, Section 4 will be revised to include a discussion of PWSCC and its effect on those components. Internals components for which PWSCC is potentially significant will be evaluated in Section 5, in terms of existing ASME Section XI inservice visual examination (VT-3) requirements.	It is believed that the response to this NRC comment fully resolves the concerns raised by the NRC. NUMARC requests that the NRC consider this comment "closed" based upon the response See Comment G-17 for additional information.

PWR INTERNALS IR: RESPONSE TO NRC COMMENTS

NRC COMMENT #	NRC COMMENT	INDUSTRY RESPONSE	STATUS	PROPOSED IR REVISION	REMARKS
S-19	Page 4-11 and 4-12: Stress corrosion cracking of a large number of components has been eliminated from further consideration for license renewal apparently based on operating experience with these components for the last 20-30 years. In order to make a case for license renewal, the IR should demonstrate that these components will not be affected by SCC through the license renewal period.	Because there has been 20-30 years of PWR internals operating experience without the effects of SCC we expect similar behavior during the license renewal period. Laboratory tests to simulate the PWR environment for durations of the same order of magnitude are unnecessary and impractical. Operating experience coupled with maintenance of the primary coolant chemistry within the limits specified throughout the current licensing period will provide adequate SCC-management of internals components. There is not any known data which support a contrary position and , in fact, much of the work performed by GE to justify (HWC) hydrogen water chemistry has also demonstrated the importance of low oxygen, low halogen water.	Open	Section 4.0 of the IR will be modified to more explicitly define the primary coolant chemistry limits and the significance of maintaining these limits throughout the license renewal period.	It is believed that the response to this NRC comment fully resolves the concerns raised by the NRC. NUMARC requests that the NRC consider this comment "closed" based upon the response. Also, see comments G-l6, G-19 and S-16.
S-20	Section 4.3: Irradiation-Assisted Stress Corrosion Cracking (IASCC): This section indicates that IASCC is not an age-related degradation mechanism for the license renewal period because of the relatively benign environment of PWR coolant, which incorporates hydrogen overpressure. This conclusion needs to be justified.	Irradiation-assisted stress corrosion cracking (IASCC) is determined to be insignificant for PWR internals on the basis of a favorable combination of these factors: 1. the relatively benign PWR coolant environment, including the hydrogen overpressure; 2. The relatively low stress levels seen by internals during normal operation;	Open	Section 4.3 the IR will be revised to clarify of the criteria that must be satisfied in order for a license renewal applicant to take credit for the generic conclusion.	It is believed that the response to this NRC comment fully resolves the concerns raised by the NRC. NUMARC requests that the NRC consider this comment "closed" based upon the response.

NRC COMMENT #	NRC COMMENT	INDUSTRY RESPONSE	STATUS	PROPOSED IR REVISION	REMARKS
		3. the residual material ductility and fracture toughness following radiation exposure. For low stresses (<10 ksi), fluence up to $5x10^{20}n/cm^2$ is acceptable in a controlled PWR coolant environment. For higher stresses (>10 ksi), the fluence should be limited to $1x10^{20}n/cm^2$. If these thresholds are exceeded, a plant-specific fracture mechanics and flaw tolerance evaluation is needed to determine a surface, visual or volumetric inspection requirement.			
S-21	Page 4-14, Section 4.4.2: The IR states that the reactor coolant fluid velocities are well below those which are required for erosion. Since the fluid velocity is a suspected cause for the observed thinning of thimble tubes, the IR should address this issue.	Fluid velocities do not cause erosion in flux thimble tubes but can, in a limited number of instances, cause wear through the action of flow-induced vibration. Thimble tube wear is discussed in Section 4.7 and 5.5 of the IR.		None	It is believed that the response to this NRC comment fully resolves the concerns raised by the NRC. NUMARC requests that the NRC consider this comment "closed" based upon the response. See also, the response to comment G-5
S-22	Page 4-15, Section 4.5.2: The IR states that for the austenitic stainless steels and Ni-base alloys of which most PWR internals components are constructed, creep is not a concern. However, the IR did not identify materials for the remaining components nor justify why creep is not significant for these components. This information should be provided in the IR.	The reactor internals components are constructed from austenitic stainless steels, Ni-base alloys, precipitation-hardened austenitic stainless steels, cobalt base alloys and martensitic stainless steels. There are no carbon steel PWR reactor vessel internal components. For the materials of construction, creep is not a concern as the operating temperatures are below 1000°F.	Open	A list of categories of materials used in the construction of the PWR internal components will be added to Section 3.0 of the IR. Section 4.5.2. will also be modified to include the entire list of material categories.	It is believed that the response to this NRC comment fully resolves the concerns raised by the NRC. NUMARC requests that the NRC consider this comment "closed" based upon the response. See also, the response to comment G-10.

NRC COMMENT #	NRC COMMENT	INDUSTRY RESPONSE	STATUS	PROPOSED IR REVISION	REMARKS
S-23	The IR did not address the issue of managing relaxation damage in inaccessible regions where UT detection is not possible. In addition, the reliability of the UT detection is questionable. Additional information concerning inaccessible regions and reliability of the UT detection should be provided in the IR.	Section 4.6 and 5.4 do not rely on UT techniques to address aging management of stress relaxation; consequently, there is no need to discuss reliability of UT techniques. As discussed in section 6.1.1. However, UT inspection techniques are considered to be a potentially viable method for providing aging management. Aging management programs which employ UT inspections would necessarily have to demonstrate their reliability. As discussed in the response to NRC Comment G-9, a list of components which cannot be inspected by the current and effective procedures in Section 5 can be made. Any component which could not be inspected by these methods would be referred to Section 6 for aging management.	Open	Proposed changes in response to comment G-9 serve to address this comment.	It is believed that the response to this NRC comment fully resolves the concerns raised by the NRC. NUMARC requests that the NRC consider this comment "closed" based upon the response. See also, the response to comments G-9 and G-10.

PWR INTERNALS IR: RESPONSE TO NRC COMMENTS

NRC COMMENT #	NRC COMMENT	INDUSTRY RESPONSE	STATUS	PROPOSED IR REVISION	REMARKS
S-24	Section 4.6 - Stress Relaxation		Open	None	It is believed that the response to this NRC comment fully resolves the concerns raised by the NRC. NUMARC requests that the NRC consider this comment "closed" based upon the response. See also, the response to comments G-9, G-10 and S-23.
	(a) This section cites eight references (Reference 4-21 through 4-28) in its discussion on the significance of stress relaxation to reactor internal components. Provide copies of those references, which contain stress relaxation information relative to Babcock and Wilcox and Westinghouse Baffle-Former Assembly Bolts and Combustion Engineering Core Shroud Assembly Bolts.	(a) These requested references will be provided under a separate cover.			
	(b) To assist the staff in its review of this issue, provide more detailed drawings or sketches of the Babcock and Wilcox and Westinghouse Baffle-Former assembly and Combustion Engineering Core Shroud Assembly.	(b) The IR recognizes stress relaxation as a Section 6 aging management issue for Babcock and Wilcox and Westinghouse baffle former assembly and combustion engineering core shroud assembly. As discussed in Comment G-9 and S-23, stress relaxation is a Section 6 aging management concern only for components which cannot be inspected by the current and effective inspection methods discussed in Section 5. On a plant-specific basis, more detailed drawings or sketches would be made available by a license renewal applicant if necessary to support NRC review of a license renewal application.			

NRC COMMENT #	NRC COMMENT	INDUSTRY RESPONSE	STATUS	PROPOSED IR REVISION	REMARKS
S-25	Page 4-23, Section 4.8: The interaction effect between thermal aging and irradiation should be addressed in the IR.	Synergistic effects have been addressed in the IRs to the extent that such effects have been observed in actual operation, relevant laboratory data, or related experience. Not all combinations of age-related degradation meet this criterion. The precipitation of chromium-rich delta ferrite and the generation of interstitial vacancies would not seem to be a likely combination that would meet this criterion.	Open	None	It is believed that the response to this NRC comment fully resolves the concerns raised by the NRC. NUMARC requests that the NRC consider this comment "closed" based upon the response.
S-26	This section indicates that mechanical properties of austenitic stainless steel castings with delta ferrite content below 20% will not be degraded significantly by thermal aging. This statement is not true based on data in NUREG/CR-5385, ANL-89-17 and a paper presented by Slams et al. at SMIRT post conference seminar 6 on Assuring Structural Integrity of Steel Reactor Pressure Boundary Component, August 1983, Monterey, California. Provide justification for the statement.	The selection of 20% delta ferrite as the threshold for potential significance of thermal aging embrittlement to cast austenitic stainless steel components was based upon a review of the existing literature, as represented by IR References 4-29, 4-30, 4-31, 4-32 and 4-33. IR Reference 4-33 compiled a relatively complete set of references c[illegible] period up to its publicatio[illegible] Since that time, O. K. Chopra and A. Sather, "Initial Assessment of the Mechanisms and Significance of Low-Temperature Embrittlement of Cast Stainless Steels in LWR Systems," NUREG/CR-5385 (ANL-89/17), was published in August 1990. Those data were available in earlier papers presented by the authors, and have been reviewed and reconciled with the industry position.	Open	NUREG/CR-5385 will be added to the reference list at the end of Section 4 and will be cited in the text of the IR.	It is believed that the response to this NRC comment fully resolves the concerns raised by the NRC. NUMARC requests that the NRC consider this comment "closed" based upon the response.

NRC COMMENT #	NRC COMMENT	INDUSTRY RESPONSE	STATUS	PROPOSED IR REVISION	REMARKS
		The data must be interpreted with some care since a "lower-bound" impact energy evaluation can be contaminated by data obtained from specimens artificially aged at temperatures well outside the range of PWR reactor coolant system operating temperatures." What NUREG/CR-5385 does not say, but should is that 400°C aging is somewhat suspect, as well. If Figures 13-15 of NUREG/CR-5385 are examined carefully, without the 450°C data and with a suspicious look at the 400°C data, the 16% delta ferrite, 18% delta ferrite, and 14% delta ferrite fall into one category; the 21% delta ferrite, 23% delta ferrite and 23% delta ferrite fall into another category. In this case, lower-bound impact energies in the former category are of the order of 50 ft-lb, while the lower-bound impact energies in the latter category are of the order of 25 ft-lb. The importance of the delta ferrite is further illuminated by Equation (6) of NUREG/CR-5385, which provides a correlation of minimum room temperature impact energy and chemistry/microstructure. The delta ferrite appears as a squared entity, dominating the correlation. The correlation also shows a weak dependence on carbon and molybdenum content, penalizing CR-8 (slightly) and CR-8M (moderately) relatively to CR-3.			

NRC COMMENT #	NRC COMMENT	INDUSTRY RESPONSE	STATUS	PROPOSED IR REVISION	REMARKS
		The threshold value of 20% delta ferrite is felt to be a conservative representation of an upper-shelf impact energy of 50ft-lb. In fact, from Figures 19-24 of NUREG/CR-5385, again removing the 450°C aged data and looking suspiciously at the 400°C data, none of the heats would yield upper-shelf impact energies below 50 ft-lb. The industry is concerned, however, with the reductions in upper-shelf energy found for delta ferrite contents above 20% (see Figures 19(a), 19(b), 20(a), 20(b), 21(a) and 21(b); compare these with Figures 20(d) and 32(d)). The upper-shelf energy for the lower delta ferrite content heats confirms the 60 ft-lb estimate. Therefore, the industry has determined that the 20% delta ferrite threshold represents a sound basis for further evaluation of cast austenitic stainless steel components subject to potential thermal aging embrittlement.			
S-27	Section 4.10 - Fatigue: There is an ongoing generic discussion on fatigue evaluation for license renewal between NUMARC and the staff. Until an agreement(s) is reached, the fatigue issues are unresolved.	In spite of the ongoing discussions between the industry and NRC staff concerning fatigue, efforts should be made to reach agreements-in-principle on fatigue issues that are amenable to resolution.	Open	None	It is believed that the response to this NRC comment fully resolves the concerns raised by the NRC. NUMARC requests that the NRC consider this comment "closed" based upon the response.

PWR INTERNALS IR: RESPONSE TO NRC COMMENTS

NRC COMMENT #	NRC COMMENT	INDUSTRY RESPONSE	STATUS	PROPOSED IR REVISION	REMARKS
S-28	Page 4-31: The IR should provide technical information in support of the newly calculated usage factors.	The summary discussion of the fatigue reanalysis performed by Babcock and Wilcox for their internals, on Page 4-30 and 4-31 of the report, is supported by IR References 4-3, 4-37 and 4-38. The most directly relevant reference for the recalculated fatigue usage factors is IR Reference 4-3, which will be supplied by under separate cover for staff review.	Open	As identified on Page 4-31 of the IR, the supporting technical information for the newly calculated usage factors is contained in Reference 4-3, Project Topical Report: Oconee Nuclear Power Station Unit 1 Reactor Vessel Internals Life Extension Project. The reference document number provided in the IR was Draft EPRI Report RP 2643-18. An alternate reference document is the original report written by Babcock and Wilcox, BAW-2060, October 1988. Both documents are identical. The IR reference document number, for Reference 4-3, will be revised to BAW-2060, October 1988, in place of the EPRI draft document number.	It is believed that the response to this NRC comment fully resolves the concerns raised by the NRC. NUMARC requests that the NRC consider this comment "closed" based upon the response. This is the same data which industry shared with the NRC during the fatigue discussions on November 14, 1990.
S-29	Page 4-[illegible] Table 4-1(a): Note 2 should be applied [illegible] lower support columns instead of fuel pin.	The reviewer is correct. This was a typo and the IR will be corrected.	Closed	Make correction on page 4-36 in the IRs.	It is believed that the response to this NRC comment fully resolves the concerns raised by the NRC. NUMARC requests that the NRC consider this comment "closed" based upon the response.

NRC COMMENT #	NRC COMMENT	INDUSTRY RESPONSE	STATUS	PROPOSED IR REVISION	REMARKS
S-30	Page 4-35 and 4-27: The IR should clarify that absence of an "S" (i.e., a potentially significant aging issue) in the degradation Table should not eliminate future ISI of that component for that specific form of degradation. In addition, radiation embrittlement should be added in this Table.	We cannot agree. For items that do not contain an "S", no additional ISI is stipulated by the IR beyond that required by current codes, standards, regulations and other commitments made by the license renewal applicant. There is no intent on the part of the IR to supersede these current commitments; instead, the IR limits the scope of license renewal assessment relative to these commitments. Neutron irradiation embrittlement was determined to be insignificant for all PWR internals, and thus does not need to be identified in Tables 4-1(a), 4-1(b) and 4-1(c).	Open	None	It is believed that the response to this NRC comment fully resolves the concerns raised by the NRC. NUMARC requests that the NRC consider this comment "closed" based upon the response.

PWR INTERNALS IR: RESPONSE TO NRC COMMENTS

NRC COMMENT #	NRC COMMENT	INDUSTRY RESPONSE	STATUS	PROPOSED IR REVISION	REMARKS
S-31	Section 5.1 Evaluation Method: This section appears to imply that a fracture mechanics analysis may be used in lieu of fatigue usage factor analysis. We do not agree with the alternate method, because of uncertainties in flaw growth predictions. The ASME Code recognizes this limitation and recommends that these flaws should receive augmented inspection in accordance with IWB-2420 of ASME Code Section XI. The IR should be revised to comply with the ASME Code or additional justification is needed.	The issue of effective programs for managing potentially significant fatigue damage has been the subject of ongoing discussions between NUMARC/industry and the staff. The industry position, as articulated in a meeting with the staff on July 16, 1991, is that at least two equivalent paths for evaluating and managing fatigue damage have been demonstrated to be effective -- fatigue analysis/reanalysis, using the procedures of the ASME Code Section III (in this case, Subsection NG), or inservice examination, using the procedures of the ASME Code Section XI (in this case, Subsection IWB). Since the applicable inservice examination requirements for reactor internals (see Table IWB-2500-1, Examination Category B-N-3) emphasize visual (VT-3) examination of accessible surfaces of components that have been removed from the reactor vessel, the industry position is that it would be prudent to supplement the inservice examination alternative with flaw tolerance evaluations, following the general outline of the evaluation steps in ASME Nuclear Code Case N-481. These steps include the postulation of a reference flaw that would be assured of detection, the location of this reference flaw in the	Open	Section 5.2 of the report will be modified to clarify the two alternatives for evaluation of potentially significant fatigue damage. The use of flaw tolerance (fracture mechanics) analysis will be specifically identified as supplementary to visual inservice examination.	It is believed that the response to this NRC comment fully resolves the concerns raised by the NRC. NUMARC requests that the NRC consider this comment "closed" based upon the response.

NRC COMMENT #	NRC COMMENT	INDUSTRY RESPONSE	STATUS	PROPOSED IR REVISION	REMARKS
		worst position and orientation relative to applied stresses, and adequate demonstration that flaw growth and flaw instability do not compromise component integrity. Material properties used in the evaluation would have to be justified by the applicant on a component-specific basis. Therefore, it is not intended that fracture mechanics analysis alone is an alternative to fatigue usage factor evaluation; instead, the flaw tolerance (fracture mechanics) procedures are intended to supplement visual inservice examination requirement.			

PWR INTERNALS IR: RESPONSE TO NRC COMMENTS

NRC COMMENT #	NRC COMMENT	INDUSTRY RESPONSE	STATUS	PROPOSED IR REVISION	REMARKS
S-32	Pages 5-6 through 5-11 and 6-13: (a) The role of the PWR environment including irradiation on fatigue crack growth was apparently not evaluated and should be considered in the IR. (b) The IR states on Page 5-11 that the scaling parameter Co in the crack growth rate formula is not substantially impacted by the PWR water environment. Since according to ASME PVP 1986 Code, the value of Co in the PWR environment could be twice that in air, the IR should be more specific about its recommended value of Co.	The following will be added to the last paragraph of Section 5.1.2.3: "However, Appendix C of Section XI of the 1989 ASME Code does not provide specific guidance on crack growth rates in water environments. Rather, the appendix states that 'Reference fatigue crack growth rates for austenitic stainless steels exposed to water environments are in the course of preparation.' While there have been estimates of the influence of PWR and BWR water environment effects on stainless steel base and weld metal, ranging from a factor of 2 to 10 higher in crack growth rate than rates in air, the values used in license renewal applications should be justified on a plant-specific basis. For most situations, fatigue crack growth rates in PWR water environments are expected to be minimal. Similarly, it has been demonstrated (see Reference) that fast neutron irradiation has no significant effect on the fatigue crack growth of austenitic stainless steels for fluence levels of interest." Reference: L.A. James, "Fatigue-Crack Propigation in Austenitic Stainless Steels," Atomic Energy Review, Vol. 14, No 1, pp 37-86, 1976.	Open	Implement the response indicated above.	It is believed that the response to this NRC comment fully resolves the concerns raised by the NRC. NUMARC requests that the NRC consider this comment "closed" based upon the response.

PWR INTERNALS IR: RESPONSE TO NRC COMMENTS

NRC COMMENT #	NRC COMMENT	INDUSTRY RESPONSE	STATUS	PROPOSED IR REVISION	REMARKS
S-33	Section 5.1.2.2 - Fatigue Re-evaluation This section cites Reference 5-3 as an example of the type of approach that could be used to re-evaluate certain fatigue analyses by using less conservative assumptions than those used in the original analyses. Reference 5-3 is the same as Reference 4-3 which is one of the documents requested in comment No. 25(a) above. There is also ongoing generic discussion on the fatigue evaluation between NUMARC and the staff. Until an agreement is reached, this issue will remain unresolved.	Reference 5-3, which is identical to Reference 4-3, will be provided under separate cover, as indicated in the response to Comment S-28. In spite of the ongoing discussions between the industry/NUMARC and the staff on fatigue, efforts should be made to reach agreement on those fatigue issues amenable to resolution.	Open	None	It is believed that the response to this NRC comment fully resolves the concerns raised by the NRC. NUMARC requests that the NRC consider this comment "closed" based upon the response.
S-34	Section 5.1.2.3 Fatigue Crack Growth Evaluation: This section indicates that fatigue crack growth may be calculated based on a scaling parameter for stainless steel in air. This section also indicates that the PWR water environment does not substantially impact the scaling factor. NUMARC should provide more information including test data to support this statement.	The ASME Code Section XI, Subsection IWB, Appendix C and, in particular, Fig. C-3210-1, show that R ratios and temperatures can be accommodated by scaling of the coefficient C_0. The effects of PWR water environment would be expected to be treated in a similar manner, except that the exponent may be affected, as well (see Fig. A-4300-2). However, the justification of fatigue crack growth rates should be a plant-specific consideration in the absence of ASME Code guidance. We believe the response to comment S-32 also addresses this comment.	Open	See the proposed revision to Comment S-32.	It is believed that the response to this NRC comment fully resolves the concerns raised by the NRC. NUMARC requests that the NRC consider this comment "closed" based upon the response.

PWR INTERNALS IR: RESPONSE TO NRC COMMENTS

NRC COMMENT #	NRC COMMENT	INDUSTRY RESPONSE	STATUS	PROPOSED IR REVISION	REMARKS
S-35	Page 5-16: The mechanically redundant systems is discussed in Section 5.1.3 instead of Section 5.1.2.	We agree. This typographical error will be corrected.		Change p. 5-16, paragraph 3 form "(Section 5.1.2)" to "(Section 5.1.3.)"	It is believed that the response to this NRC comment fully resolves the concerns raised by the NRC. NUMARC requests that the NRC consider this comment "closed" based upon the response.
S-36	Section 5.2 - Internals Fatigue Evaluations This section identifies the following two established methodologies that are available to verify the adequacy of the fatigue design basis throughout the license renewal term: (a) Reanalysis in accordance with the procedures described in Section 5.1.2.2; or (b) Inservice Inspection (ISI) in accordance with ASME Section XI. As mentioned in comment No. 31 above, the staff has not yet endorsed Option 1. With respect to Option 2, the staff does not agree that ISI in accordance with the provisions of ASME Section XI is a viable alternative to plant specific analysis which calculates cumulative usage factors. Inspection cannot substitute for analyses, e.g. ISI will not detect fatigue effects that had not yet resulted in crack initiation. Once a crack initiates, rapid propagation may lead to failure between ISI intervals. Historically, ISI has not detected reactor coolant pressure boundary failures before they have occurred. This section should be revised to reflect any agreement(s) to be reached in the ongoing discussion on fatigue evaluation.	We cannot agree with the reviewer's comments. There is no prohibition against fatigue reanalysis during the current license therm, provided that such analysis is carried out in accordance with ASME Code Section III procedures. Such re-analysis are performed on a routine basis by current licenses. There is no reason to suspect that fatigue reanalysis is unacceptable for the license renewal term. ISI is not required to detect fatigue crack initiation, just the existence of macroscopic fatigue cracks of a size that approaches acceptance criteria limits of Section XI. The industry recognizes that substitution of ISI for fatigue analysis/reanalysis will not be permitted unless the component location is sufficiently tolerant to such macroscopic flaws to justify the periodic ISI interval. It is the industry's position that ISI has performed well in detecting and sizing flaws that threaten the integrity of the reactor coolant pressure boundary.	Open	The IR will be revised to reflect agreements reached with the staff of fatigue evaluation.	It is believed that the response to this NRC comment fully resolves the concerns raised by the NRC. NUMARC requests that the NRC consider this comment "closed" based upon the response.

NRC COMMENT #	NRC COMMENT	INDUSTRY RESPONSE	STATUS	PROPOSED IR REVISION	REMARKS
S-37	Section 5.4 - Internals Stress Relaxation Evaluations This section indicates that stress relaxation is not a significant age-related degradation mechanism for the components listed on page 5-18 because of redundancy and periodic visual inspection. It is likely that thermal and irradiation conditions would affect the preload of all bolts or tie rods in the components. Hence, redundancy is not an acceptable justification. Furthermore, visual inspection is not designed to detect stress relaxation. Since many of these components are subject to irradiation embrittlement and stress corrosion cracking, they may be susceptible to multiple and rapid failure. Therefore, all the components listed on page 5-18 need age-related degradation management.	The issue of common-mode failure was addressed in the response to Comment G-20. Thermal and radiation conditions are not likely to affect the pre-load of all bolts or tie rods in the components identically, since the response of each bolt or tie rod depends on chemistry and mechanical factors that vary, in some cases considerably. The variation in bolt hardness and bolt preload alone is sufficient to preclude any common mode failure concerns, unless the loading is extremely severe. The VT-3 visual inspection requirement of the ASME Code Section XI (see IWB-3520.2, e.g.) lists "loose, missing, cracked, or fractured parts, bolting, or fasteners". With consideration of preload variability and other variation taken into account, the relevant conditions of IWB-3520.2 are sufficient to detect and manage potentially significant stress relaxation. We do not agree that the failures will be multiple and rapid.	Open	None	It is believed that the response to this NRC comment fully resolves the concerns raised by the NRC. NUMARC requests that the NRC consider this comment "closed" based upon the response.

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NRC COMMENT #	NRC COMMENT	INDUSTRY RESPONSE	STATUS	PROPOSED IR REVISION	REMARKS
S-38	Section 5.4 - Internals Stress Relaxation Evaluations In this section, it is concluded that all bolts and fuel pins with the exception of Baffle-Former and Core Shroud Assembly Bolts are eliminated from further evaluations based on the following: (a) Bolts and pins represent mechanically redundant systems. (b) Inservice inspection will detect damage and retightening or replacement of the bolts and pins that are found to be damaged is a proven technology. It is not clear how the above conclusions apply to fuel pins. Provide more detailed information to justify the elimination of fuel pins from further evaluation.	The fuel pins referred to are those attached to the core support plate to restrain lateral motion of the fuel assemblies. There is more than one fuel pin per fuel assembly; consequently, there is redundancy. Furthermore, fuel pins are accessible components and any stress relaxation sufficient to cause loosening will be evident by fret marks. Any fuel pins which are damaged will be even more readily detectable. Hence, the IR will use visual inspecion as an established program for the evaluation of fuel pins. See also the response to NRC Comment G-6 relative to stress relaxation detection.	Open	Revise the IR to close the stress relaxation of the fuel pins in Section 5 based upon visual inspection requirements.	It is believed that the response to this NRC comment fully resolves the concerns raised by the NRC. NUMARC requests that the NRC consider this comment "closed" based upon the response.

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NRC COMMENT #	NRC COMMENT	INDUSTRY RESPONSE	STATUS	PROPOSED IR REVISION	REMARKS
S-39	Page 5-19, last paragraph: The IR states that wear between two surfaces subject to relative motion is readily detectable by visual inspection long before the effects of wear begin to comprise the structural integrity or function of the component. The current practice of inspection once every 10 years does not make wear readily detectable long before its effects begin to compromise its integrity. The IR should consider an augmented inspection program for all susceptible safety-related components as a requirement for license renewal.	Current ASME Code Section XI inspection requirements (e.g., examination Category B-N-3) are effective in detecting wear on a timely basis. However, such requirements have evolved through the consensus process over the years, and have been evaluated for any time dependencies that might preclude their effectiveness for either the current of the license renewal term. Wear of surfaces is listed as a relevant condition for Examination Category B-N-3 in IWB-3520.2. Industry has determined that this established program is currently effective, and we can find no reason to suspect that wear will accelerate during the license renewal period.	Open	Section 5.5 of the IR will be revised to discuss the relevant conditions of the VT-3 inspection required by Examination Category B-N-3.	It is believed that the response to this NRC comment fully resolves the concerns raised by the NRC. NUMARC requests that the NRC consider this comment "closed" based upon the response.

NRC COMMENT #	NRC COMMENT	INDUSTRY RESPONSE	STATUS	PROPOSED IR REVISION	REMARKS
S-40	Section 5.6 Internals Thermal Aging Evaluation: This section needs to be rewritten because it indicates thermal aging effects have not been observed in cast austenitic stainless steel components and inservice inspection has not identified any cracking which was attributed to thermal aging. Thermal aging does not cause cracking. It reduces the load carrying capability of cracked components because the material has lower ductility. To ensure that the cast austenitic stainless steel components do not significantly lose their load carrying capability, all the components must be periodically inspected to detect cracks. See General Comment No. 15.	Section 5.6 of the IR will be revised to more accurately scribe the effects of potentially significant thermal aging embrittlement on PWR internals, and to describe the elements of effective programs for managing these effects. It is recognized that thermal aging embrittlement of cast austenitic stainless steel components does not, in itself, cause cracking; however, the presence of defects and/or flaws, when combined with the effects of thermal aging embrittlement, is a concern.	Open	The paragraph at the bottom of p. 5-21 will be revised to read as follows: "Lower support columns and upper support columns in Westinghouse plants, and core support columns in Combustion Engineering plants, are under compressive loads during normal operation. Furthermore, these components are subject only to coolant inlet temperatures which significantly reduces the extent of thermal aging. Extension of any preexisting flaws or defects, as the result of aging-enhanced crack growth rates, or flaw instability, as the result of aging-reduced fracture toughness, is mitigated by the absence of tensile stresses. The combination of relatively infrequent inservice visual examination and flaw tolerance evaluation provides the basis for managing the effects of thermal aging embrittlement for these components. The flaw tolerance methodology can be adapted from ASME Code Section XI, Appendix C, with a postulated reference flaw substituted for flaws detected and sized during inservice examination. The orientation and size of the reference flaw must be justified in terms of the component loadings and inservice examination requirements."	It is believed that the response to this NRC comment fully resolves the concerns raised by the NRC. NUMARC requests that the NRC consider this comment "closed" based upon the response.

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NRC COMMENT #	NRC COMMENT	INDUSTRY RESPONSE	STATUS	PROPOSED IR REVISION	REMARKS
S-41	Pages 5-20 and 5-21: Evaluation of thermal aging effects should include the interaction with irradiation where applicable. the IR should consider such interaction especially since visual inspection to detect flaws that may have occurred would not be adequate if slow cracking caused by thermal aging is untimely.	The potential for increased crack growth rates in cast austenitic stainless steel components as the result of a combination of thermal aging embrittlement and neutron irradiation embrittlement has not been identified as a credible synergistic mechanism.	Open	None	It is believed that the response to this NRC comment fully resolves the concerns raised by the NRC. NUMARC requests that the NRC consider this comment "closed" based upon the response. This comment is similar to Comment S-25.

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NRC COMMENT #	NRC COMMENT	INDUSTRY RESPONSE	STATUS	PROPOSED IR REVISION	REMARKS
S-42	Pages 5-20 to 5-23, Section 5-6. The 20 percent ferrite criterion is an inadequate tool to screen cast stainless steel compositions which may be subject to significant embrittlement by thermal aging. An appropriate fracture mechanics analysis is a sound tool for the management of this problem. However, scoping calculations are needed to demonstrate that VT-3 inspections or other inspections are adequate to detect flaws less that the critical size and to ensure that inspection intervals are adequate. As noted previously, we do not believe that VT-3 can reliably detect tight cracks.	The 20% delta ferrite screening threshold for potential significance of thermal aging embrittlement of cast austenitic stainless steel internals has been discussed extensively in the response to Comment S-26. We agree with the reviewer that a fracture mechanics analysis, such as that described in Section 5.6 of the IR, is effective for management of the potential problem. At issue is the effectiveness of the VT-3 visual inspection to detect cracking prior to reaching a critical size. The industry position is based on the requirements of Examination Category B-N-3, and the definition of relevant conditions cited in IWB-3520.2. In particular, "loose, missing, cracked, or fractured parts" is identified as a relevant condition. The consensus code bodies, in particular the group working on subsection IWG may wish to consider the the use of VT-1. Until such time as the consensus bodies complete their deliberations, the industry is satisfied with detecting cracks with VT-3, followed by supplementary inspection when relevant conditions are found.	Open	None	The first part of the comment is similar to Comment S-26.

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NRC COMMENT #	NRC COMMENT	INDUSTRY RESPONSE	STATUS	PROPOSED IR REVISION	REMARKS
S-43	Page 6-2: It is not clear from the discussions presented in the IR how the proposed evaluation methods assure detection of stress relaxation.	The report mentions, on Page 6-2, that "A potential also exists for applying UT inspection techniques to bolt stress relaxation, given appropriate baseline data on as-built bolt dimensions." The report does not elaborate, because the procedure discussed is still in the research and development stage. The technique is based on the relatively small changes in propagation velocity of ultrasonic waves along the length of the bolt. If extremely accurate baseline data is available, changes in bolt preload caused by stress relaxation could be detected. The current stage of development of the method does not warrant more detailed discussion.	Open	None	It is believed that the response to this NRC comment fully resolves the concerns raised by the NRC. NUMARC requests that the NRC consider this comment "closed" based upon the response.