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Comment On: NRC-2012-0070-0001
Updated Aging Management Criteria for Reactor Vessel Internal Components of Pressurized Water Reactors

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Submitter Information

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General Comment

Comments on LR-ISG-2011-04 are being submitted on behalf of the Electric Power Research Institute Materials Reliability Program and the Pressurized Water Reactor Owners Group Materials Subcommittee. The comments are included in the letter uploaded below. If you have any questions, please contact Jean Smith at jmsmith@epri.com.

Attachments

MRP 2012-027 with Attachment

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(via email)

May 21, 2012

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Subject: Industry Comments Pertaining to Draft License Renewal Interim Staff Guidance "Updated Aging Management Criteria for PWR Reactor Vessel Internal Components" (LR-ISG-2011-04), Docket ID NRC-2012-0070

Reference: Federal Register Volume 77, Number 54, March 20, 2012, pages 16270 –16271.

In the referenced document, the U.S. Nuclear Regulatory Commission (U.S. NRC) requested public comment on the subject draft License Renewal Interim Staff Guidance (LR-ISG). The draft LR-ISG updates the guidance in the Standard Review Plan for Review of License Renewal Application for Nuclear Power Plants (SRP-LR) and Generic Aging Lessons Learned (GALL) Report for the aging management of stainless steel structures and components exposed to treated borated water.

The comments in the Attachment have been jointly prepared by the Electric Power Research Institute Materials Reliability Program (EPRI-MRP) and the Pressurized Water Reactor Owners Group Materials Subcommittee (PWROG-MSC). These two industry groups have worked closely with the Nuclear Energy Institute's (NEI's) License Renewal Task Force to ensure consistency and alignment in representing the industry's position. Further, EPRI-MRP and the PWROG-MSC endorse the comments submitted by NEI on behalf of the industry. The content of the comments is based on feedback received by the industry groups from the U.S. NRC during the Public Meeting held on April 26, 2012 in Rockville, MD.

The comments submitted by EPRI-MRP, the PWROG-MSC, and NEI are extensive and involve complex issues. EPRI and the PWROG, along with NEI, respectfully request a follow-up meeting with the NRC staff to discuss resolution of the comments and, if appropriate, an additional comment period.

The industry appreciates the opportunity to be involved in the review and comment process for the ISG and respectfully requests that you consider for incorporation the comments as stated in the attachment. If you should have any questions concerning this letter, please contact Jean Smith (JMSmith@epri.com) at 650-855-8775.

Sincerely,

Timothy G. Wells
Southern Nuclear Company
Chairman, EPRI-MRP

Edison Fernandez
Arizona Public Service Co.
Chairman, PWROG-MSC

Attachment: Industry Comments on Draft License Renewal Interim Staff Guidance (LR-ISG) 2011-04

cc: Anne Demma, EPRI
Craig Harrington, EPRI
Jim Molkenthin, PWROG
Mark Richter, NEI
William Sims, Entergy
Sheldon Stuchell, U.S. NRC

ATTACHMENT

INDUSTRY COMMENTS ON
DRAFT LICENSE RENEWAL INTERIM STAFF GUIDANCE (LR-ISG) 2011-04

NON-PROPRIETARY

EPRI-MRP/PWROG-MSC Comment 1

EPRI document MRP-227-A *Materials Reliability Program: Pressurized Water Reactor Internals Inspection and Evaluation Guidelines*, which incorporates the NRC Safety Evaluation Report (SER) additions, should be the source document for an aging management program for pressurized water reactor (PWR) reactor internals. Further, the component-specific line items for aging management review (AMR) could be eliminated, or at least significantly reduced, to leave only those items for which licensee-specific actions are defined in MRP-227-A.

When NUREG-1800 and NUREG-1801 were being developed, MRP-227 Rev. 0 was not available. As a result, guidelines were needed to ensure thorough and adequate license renewal reviews of PWR reactor internals were performed; therefore, component-specific line items were included in NUREG-1800 and NUREG-1801. As MRP-227 was developed and licensees began to develop their aging management programs, it was important to have guidance for AMRs to ensure consistent reviews of a license renewal application submittal. When the current revision of the GALL was published, MRP-227 Rev. 0 had been submitted to the NRC but had not received final NRC approval. Thus, the NUREG-1800 and NUREG-1801 revisions retained the component-specific line items for reactor internals. The NRC and industry cooperated on the reconciliation of those line items with MRP-227 Rev. 0 and supporting technical reports.

The NRC reviewed and approved with limitations MRP-227 Rev. 0, and subsequently, MRP-227-A was published to incorporate the SER additions. All needed actions for licensees are contained in MRP-227-A. As a result, it is appropriate for the NRC to review a licensee's PWR reactor internals aging management program against the criteria contained in MRP-227-A. As such, it is not necessary to include all the details currently in NUREG-1800 and NUREG-1801 regarding PWR reactor internals, and instead, only a reference to MRP-227-A should be made. Outlining the requirements for reactor internals in the ISG may lead to confusion with respect to the implementation of duplicate requirements, may cause undue NRC staff burden reconciling the documents each time MRP-227 is revised by the industry, and will likely lead to human errors in document alignment through future revisions.

EPRI-MRP/PWROG-MSC Comment 2

In section 3.1.2.2.9.A.1 of Appendix A of LR-ISG 2011-04 it is stated that:

Applicants applying for license renewal of their PWR facilities are requested to provide their specific responses to the [applicant/licensee action items] A/LAIs on the MRP-227-A methodology in Appendix C of their [license renewal applications] LRAs, and to address the information requested in the LRA sections that respond to the specific SRP-LR further evaluation "acceptance criteria" that are based on these A/LAIs."

This statement requires that licensees include responses to applicant action items in both Appendix C of the LRA and in appropriate further evaluation sections of the LRA. This duplication of information provides no significant value to the reviewers. It is recommended that all A/LAI responses be included only in Appendix C, so they are in an easily-referenced location. Any additional discussion of the A/LAIs in the further evaluation sections of the SRP should be limited to identifying each of the items requiring responses and any details necessary to ensure responses are adequate. Any other items requiring discussion of the A/LAI responses in further evaluation sections of the LRA should be deleted or reference made to Appendix C of the LRA. This change will be consistent with the process most BWR applicants have been using in the past to address responses to A/LAIs for BWRVIP documents.

EPRI-MRP/PWROG-MSC Comment 3

In NUREG-1801 Rev.2 XI.M16A Program Description, last paragraph, as well as in ISG-LR-2011-04 Section 3.1.2.2.9.A.2, both an aging management program and an inspection plan are required to be submitted as part of an applicant's license renewal application. In Section 3.1.2.2.9.A.2 it states

A/LAI No. 8, Item 2 in Section 3.5.1of the NRC SE (Rev. 1) on MRP-227 states that each PWR applicant for renewal should submit an inspection plan for their RVI components that addresses those plant-specific A/LAIs that are applicable to the NSSS-design of their RVI components and to submit the inspection plan for NRC review and approval consistent with the current licensing basis (CLB) for the facility.

However nowhere in these two documents is there any clear guidance on the information that should be included in an inspection plan. This ambiguity could lead to applicants submitting information that might not meet NRC needs in this area.

In order to address this situation it is requested that the aging management program and inspection plan for an applicant be clearly defined. It is proposed that the aging management program address the 10 program element recommendations for PWR RVI components in GALL AMP XI.M16A, PWR Vessel Internals (AMP XI.M16A in NUREG-1801, Revision 2). The inspection plan could be included within a program (i.e. a program/plan) or be a separate document if submitted with a license renewal application. The inspection plan should address each of the following elements regarding the inspections that are part of the aging management program.

- Identification of items requiring inspection
- Specification of the type of examination appropriate for each item
- Schedule of initial inspection and frequency of subsequent inspections
- Criteria for sampling and coverage
- Criteria for expansion of scope if unanticipated indications are found
- Inspection acceptance criteria

The industry believes these elements are satisfied by the applicable line items from Tables 4-1 through 4-9 and Tables 5-1 through 5-3 of MRP-227-A. The inspection plan submitted as part of a license renewal application (LRA) should be included in Appendix C of the LRA along with the responses to the A/LAI items since it is a requirement of A/LAI No. 8.

EPRI-MRP/PWROG-MSC Comment 4

Appendix A, Section 1, Staff Proposed Revision to AMP XI.M16A, "PWR Vessel Internals," in the Generic Aging Lesson Learned Report, Revision 2 (GALL Report, Revision 2), under Item 4, Detection of Aging Effects, stipulates that visual (VT-3) examination methods *may be applied for the detection of cracking if justified in accordance with the NRC's "further evaluation" criteria in SRP-LR Section 3.1.2.2.9.A.7, which provides the NRC's further evaluation "acceptance criteria" recommendations on this matter.*" The second paragraph of Appendix A, Section 2, Acceptance Criteria Item 3.1.2.2.9.A.7 (Use of VT-3 Visual Inspection Techniques for Detection of Cracking) refers to this limitation in AMP XI.M16A, in two ways. First, for detection of cracking in non-redundant RVI components, tolerance to "easily detected large flaws" must be demonstrated for the non-redundant component, even for reduced fracture toughness conditions (e.g., significant neutron irradiation exposure). Second, for redundant RVI components (e.g., baffle-former bolt assemblies), the applicant must identify the functional tolerance of the redundant assembly to some number of failed assembly elements (i.e., bolts). The third paragraph again refers to these justifications in order to credit VT-3 for "monitoring cracking." The fourth paragraph extends the flaw tolerance justification to non-redundant CASS, PH SS, and martensitic stainless steel components, adding a requirement for the applicant to demonstrate detection capability under reduced fracture toughness conditions, explicitly referring to reduced toughness caused by a combination of thermal aging and neutron irradiation embrittlement. The fifth paragraph summarizes the functional tolerance requirement for redundant component assemblies. In addition, corresponding references to these requirements can be found in the proposed changes to SRP-LR Table 3.0-1 (see Appendix A, Page A-30 et seq.) and in the Appendix A, Section 4 proposed AMR Item changes for GALL Section IV.B2, GALL Section IV.B3, and GALL Section IV.B4.

The stipulation of appropriate inspection methodologies for these reactor internals components has already been addressed in the review of MRP-227-A, The recommended inspection methods have already been reviewed and found to be adequate to detect the relevant conditions. The AMP attribute that is at issue is not detection of aging effects; instead, the issue is the applicant's corrective action program, and the disposition of relevant conditions through supplemental examination or engineering evaluation, both of which are outside the scope of the Mandatory or Needed requirements of MRP-227-A. Standards for engineering evaluation are addressed in Section 6 of MRP-227-A and in the methodologies described in WCAP-17096. These recommendations are based on the practice used in Section XI of the ASME code and are consistent with existing aging management programs. Further justification for the use of the VT-3 examination is not necessary and should not be required by the ISG.

It is recommended that Acceptance Criteria Item 3.1.2.2.9.A.7 (Use of VT-3 Visual Inspection Techniques for Detection of Cracking) be completely eliminated and replaced by a limited requirement to address the acceptability of VT-3 as a management approach for components that 1) were not already considered for aging management in the development of MRP-227-A, 2) are evaluated to require active aging monitoring, and 3) are non-redundant.

The justification for the industry recommendations is provided in the following paragraphs:

1. Visual (VT-3) *examinations are already accepted* in the ASME Code Section XI, Subsection IWB-3520.2, Examination Category B-N-3, for the monitoring of cracking in removable core support structures in PWRs, as illustrated by the relevant condition "loose, missing, cracked or fractured parts, bolting, or fasteners." The Examination Category B-N-3 relevant conditions are only intended to detect and report any "loose, missing, cracked or fractured parts, bolting, or fasteners," with no intention to size and compare such observations with acceptance criteria. Any reported relevant condition is unacceptable for continued service without corrective action, in accordance with IWB-3142, which means supplemental examination (e.g., VT-1 or EVT-1 examination), analytical evaluation, or repair/replacement. With the exception of more descriptive relevant conditions intended to provide additional guidance to inspectors, MRP-227-A has followed the Examination Category B-N-3 ASME Code requirements to the letter, with reporting and corrective action requirements for each and every relevant condition, and with no intent to directly size or summarily accept a detected surface-breaking discontinuity. Therefore, Acceptance Criteria Item 3.1.2.2.9.A.7 is not applicable and would only be applicable in the context of acceptance in the presence of relevant conditions, i.e., for corrective action, such as for a supplemental examination intended to length size a surface-breaking discontinuity or an analytical evaluation intended to demonstrate acceptance for continued service.

2. There is no need for any technical justification to use visual (VT-3) examination to monitor cracking in redundant RVI components, since the ASME Code Section XI, Subsection IWB-3520.2 relevant conditions "loose, missing, cracked or fractured parts, bolting or fasteners," for examination of an individual component (e.g., bolt) in a redundant assembly, plus the more descriptive relevant conditions contained in MRP-227-A, are adequate for reporting and inclusion in the applicant's corrective action program. The amount of redundancy has no bearing on the ability to detect the aging effects. Only an absence of relevant conditions is acceptable without corrective action as described in Paragraph 1 above; therefore, there is no need for its inclusion. The corrective action steps must be taken prior to returning the component or component items to service, not prior to performing the examination. The portion of Acceptance Criteria Item 3.1.2.2.9.A.7 for redundant RVI assemblies is more appropriately directed at the corrective action program, with its requirement for functional tolerance of the entire assembly. Corrective action engineering acceptance criteria do not belong in SRP-LR Section 3.1.2.2.9 and should be described within the context of the NRC staff review and acceptance of the methodologies in WCAP-17096.

3. A sound regulatory assessment of visual (VT-3) examination and justification for its use to manage various aging effects, including such effects as gross cracking and complete separation of parts, was provided in the March 28, 2011, Response to Non-Concurrence Regarding Safety Evaluation for Topical Report MRP-227, "Pressurized Water Reactor Internals Inspection and Evaluation Guidelines," by Matthew A. Mitchell, Chief, Vessels and Internals Integrity Branch, Office of Nuclear Reactor Regulation. That assessment contained several justifications, including: (i) operational experience with detecting and managing the effects of cracking; (ii) improvements in the quality of VT-3 examinations over the years as critical examination parameters such as surface cleanliness, lighting, camera speed, and character recognition height qualifications, have been refined; and (iii) evidence from studies (such as MRP-210) showing that PWR RVI components for which VT-3 examinations have been credited are very flaw tolerant, either because of the size of the component in question and the length of flaw required to begin to postulate the potential for failure, or because a group of like components is considered in which multiple like components must fail in order to compromise the functionality of the group.

4. The flaw tolerance justification requirement for non-redundant CASS, PH SS, and martensitic stainless steel components falls into a similar category, since the use of visual (VT-3) examination for detecting *surface-breaking discontinuities* for components fabricated from such materials is well established in the ASME Code Section XI. The flaw tolerance justification requirement appears to again be directed at corrective action disposition through either supplementary examination (justifying the use of a more accurate examination, such as a volumetric (UT) examination to further characterize the relevant condition) or engineering flaw tolerance evaluation. Both situations are outside the scope of MRP-227-A Mandatory and Needed requirements, with the guidance in Section 6 of MRP-227-A offered For Information Only. With the acceptance of MRP-227-A, the industry has regarded any issues concerning the adequacy of the examinations specified therein to be resolved.

EPRI-MRP/PWROG-MSC Comment 5

Appendix A – Section 2, Acceptance Criteria Item 3.1.2.2.9.A.9 (Identification of TLAAs for PWR-Design RVI Components) on Page A-20 and A-21 stipulates that, *in order to satisfy the requirements of the ASME Code , Section III, Subsections NG-2160 and NG-3121, license renewal applicants demonstrating acceptability of RVI components with design-basis cumulative usage factor (CUF) analyses that are TLAAs should include the effects of the reactor coolant system water environment in the fatigue CUF analyses.* This last sentence should be removed for the reasons provided below:

1. NG-2160 stipulates that consideration of deterioration of material caused by service is generally outside the scope of the Subsection (meaning outside the scope of Subsection NG), and instead places the responsibility for special attention to the effects of service conditions upon the properties of the material on the Owner. Consideration of an issue

does not rise to a requirement to include the effects of the reactor coolant system water environment on any fatigue CUF analyses.

2. NG-3121 notes that "the tests on which the design fatigue curves (Fig. I-9.0) are based did not include tests in the presence of corrosive environments which might accelerate fatigue failure." Again, a note to such effect does not constitute a design requirement.

3. In order for an Owner or a designer to evaluate the implications of the ASME Code "consideration" and "note" in NG-2160 and NG-3121, respectively, the December 26, 1999, Generic Safety Issue (GSI) 190 close-out memorandum from Ashok C. Thadani, Director of the Office of Regulatory Research, to William D, Travers, Executive Director for Operations, provides the technical basis for regulatory concern. The second paragraph of that memorandum provides the high-level regulatory concern, which is pipe leakage – presumably from the reactor coolant pressure boundary.

"*However, calculations including environmental effects, that were performed to support resolution of this issue, and the nature of age-related degradation indicate the potential for an* ***increase in the frequency of pipe leaks*** *as plants continue to operate.*"

The analytical support for the close-out decision is described in Attachment 1, which covers the probabilistic fatigue calculations for piping components. It should be noted that the concern is limited to pipe leakage, which is not applicable to RVI components since they do not form a portion of the reactor coolant pressure boundary and are therefore not subject to leakage.

"*However, the calculations supporting resolution of this issue, which included consideration of environmental effects, and the nature of age-related degradation* ***indicate the potential for an increase in the frequency of pipe leaks*** *as plants continue to operate.*"

Thus, fatigue issues concerning the structural integrity of the existing fleet of PWR reactor internals, which do not have a pressure boundary function, are significantly less when compared to pressure boundary components. Code fatigue analyses traditionally performed for the last generation of reactors were a demonstration to satisfy a Code minimum requirement and are commonly acknowledged to not be predictive of failure. Further, the frequency of fatigue transients in PWR reactor operation, such as heat up and reactor trips, is very low. The only fatigue failures documented in operating experience have been those events due to high cycle vibration fatigue, which is not affected by environmental effects. Therefore, evaluation of CLB TLAAs for the reactor internals should be addressed in accordance with the existing 10CFR Part 54 requirements without the need to include environmental effects.

EPRI-MRP/PWROG-MSC Comment 6

The component specific AMR items described in Appendix-A, Sections 4, 5 and 6 are based on migration from NUREG-1801. As a result the listing is more complex than the approved MRP-227-A tables. For example, there are approximately 25 items in Section 5 that classify as

"Primary" component examinations, whereas the equivalent component list in MRP-227-A contains only 13 items. The component content is very similar but the breakdown is complex. A key advantage of aligning license renewal commitments to the MRP-227-A format is to facilitate important, industry-wide program updates based on Operating Experience through the NEI 03-08 process. The alignment between MRP-227-A and NUREG-1801 is compromised by embedding item detail in the ISG format. It is recommended that NUREG-1801 refer existing AMR items to "the applicable MRP-227-A table" and retain detail only for those items which may be beyond the scope of MRP-227-A. This will significantly reduce applicant and NRC staff burden, and improve integration of evolutionary changes through the NEI 03-08 process.

EPRI-MRP/PWROG-MSC Comment 7

ISG implementation of Applicant / Licensee Action items from the MRP-227-A SER is by way of notes to AMR items listed in Sections 4, 5 and 6. This could be addressed by reference to the appropriate SER action items. It is recommended that the required evaluations would be documented in a single location specified by the ISG rather than associated with individual items. Associating these actions with each individual AMR items increases the burden for both the applicant and NRC staff reviewer.

EPRI-MRP/PWROG-MSC Comment 8

The draft ISG requires Applicants to develop and submit evaluation of inaccessible Reactor Vessel Internal components in accordance with Note 3 to Sections 4 and 5, and Note 2 to Section 6. With the exception of A/LAI #6 of the MRP-227-A SER, these evaluations have been addressed during review and approval of the Industry program. The requirement to develop, submit and review the inspection basis is unnecessary. It is recommended that this note be eliminated.

EPRI-MRP/PWROG-MSC Comment 9

MRP-227-A provides applicants with an alternative to the defined inspection requirements when plant-specific analyses of accumulated fatigue usage are performed. Applicants may choose to either inspect in accordance with the approved MRP-227-A schedules, or perform analyses. In cases where Applicants perform analyses to relax MRP-227-A requirements, those analyses would be submitted for NRC staff approval in accordance with A/LAI 8. The ISG is unclear regarding these alternatives. For example item IV.B3.RP-343 appears to require physical examinations to support acceptance of the TLAA. The industry recommends that the ISG refer to MRP-227-A and the associated A/LAI requirement discussions.

EPRI-MRP/PWROG-MSC Comment 10

Item 9.B.1 of the ISG notes that Section 3.2.5.3 of the NRC SE (Rev. 1) on MRP-227 Rev. 0 recommends that the applicant consider replacement or inspection activities with regard to the Control Rod Guide Tube (CRGT) split pins if the pins are currently fabricated with Alloy X-750 or Type 316 stainless steel material. A review of the referenced section of the SE does not reach the conclusion that this specificity of action is required; the SE requirement is to evaluate the adequacy of the plant-specific existing program to ensure that the aging degradation is adequately managed during the extended period of operation. The SE direction is on evaluation of the performance of the existing program and does not suggest that it should be changed to include inspections. Therefore the industry considers the specificity of direction provided in the SE to be sufficient and the ISG should not provide alternate direction.

EPRI-MRP/PWROG-MSC Comment 11

Section C.3, page A23 of LR-ISG 2011-04 states that per MRP-227-A, "...EVT-1 inspections of certain CE-design components would be necessary if the design basis fatigue TLAAs for the components could not demonstrate that fatigue-induced cracking would be adequately managed..." This statement does not accurately represent MRP-227-A Table 4-2, because it assumes that the fatigue evaluations required by the MRP-227-A table item already exist and are part of the current licensing basis, and therefore are formally classifiable as TLAAs. In fact, many, if not all, of the older CE design reactor internals were not qualified to the fatigue rules of ASME III, so TLAAs as defined in 10 CFR Part 54 do not exist. Further, page A24 of the draft ISG states "Otherwise, CE-design applicants for renewal are requested to credit the MRP's EVT-1 basis in MRP-227-A as the applicable aging management basis if either: (1) the CLB does not include applicable CUF or It fatigue analyses for these components;..." This statement appears to compel the applicant who does not have a current licensing basis TLAA to perform EVT-1 inspections. MRP-227-A clearly does not require inspections based solely on the lack of a current licensing basis TLAA. In fact, it only requires that a fatigue evaluation be performed to determine if a fatigue issue might exist; and if so, where would inspection be focused to manage it. The method of the fatigue evaluation was intended to be the usual engineering practice, for example by comparison of the number expected operating transient cycles to those specified by design, or by stress analysis if required.

EPRI-MRP/PWROG-MSC Comment 12

For A/LAI No. 2, when comparing the licensee renewal AMR from BAW-2248A to the tables in MRP-189, the locking devices for the vent valve were identified as a possible "Primary" component. The original vent valves located next to outlet nozzles failed due to flow induced vibration, and those valves next to the nozzles were replaced with locking devices made containing Alloy 600.

It is recommended that Table IV Reactor Vessel, Internals, and Reactor Coolant System, B4 Reactor Vessel Internals (PWR) – Babcock and Wilcox on page A-124 of LR-ISG 2011-04 be revised to include a line item addressing Alloy 600 replacement vent valve locking devices, which are subject to aging degradation due to PWSCC.

EPRI-MRP/PWROG-MSC Comment 13

In Item 8 on page A-11 of the LR-ISG, the second sentence appears to be incomplete with respect to the statement pertaining to "...confirming that the quality of inspections, flaw evaluations, and corrective actions performed under this program." It is recommended that the revised statements be reviewed for completeness.

EPRI-MRP/PWROG-MSC Comment 14

Item 3 on page A-16 of the LR-ISG should reference NRC SE Section 3.2.5.1 and not Section 3.5.1. It is recommended that this reference be revised.

EPRI-MRP/PWROG-MSC Comment 15

Item D.1 on page A-25 discusses evaluation "Acceptance Criteria" recommendations applicable to Babcock and Wilcox reactor internals. In general, A/LAI 4 is not specific relative to the wording for the manner in which the items were stress relieved, and it was stated that a "stress relief process" was used. In Item D.1, the wording used in some cases implies a "post-weld heat treatment" process. The words "stress relief process" should be used consistently without the implication of a heat treatment process only. In addition, the requirements in Item D.1 appear to go beyond the requirements of the A/LAI as it was written and approved by the MRP-227-A SER.