

April 5, 2001

Howard H. Chung
MITEC International, Inc.
1022 Revere Court
Naperville, IL 60540

SUBJECT: SUBMITTAL OF PAPER TO PVP 2001 CONFERENCE

Dear Mr. Chung:

Enclosed is the final paper "POTENTIAL ENHANCEMENTS OF CODES AND STANDARDS TO SUPPORT LICENSE RENEWAL" for Track 10, "NRC/ASME Symposium on BPVC Section XI," in ASME format. If you have any questions, please contact me by e-mail at jxd@nrc.gov or by telephone at (301) 415-1014.

Sincerely,

/RA/

Jerry Dozier, General Engineer
License Renewal and Standardization Branch
Division of Regulatory Improvement Programs
Office of Nuclear Reactor Regulation

Enclosure: As stated

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POTENTIAL ENHANCEMENTS OF CODES AND STANDARDS TO SUPPORT LICENSE RENEWAL

Jerry Dozier

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Abstract

Nuclear power plants (NPPs) were originally designed and constructed in accordance with accepted codes and standards. These codes and standards provided sufficient design conservatism to accommodate normal, upset, emergency, and faulted design loading conditions. Operating experience available today has made it possible to identify specific areas where the design conservatism has potentially eroded as NPPs age. Most of the potential degradation due to aging is mitigated, prevented, or detected with existing plant programs (e.g., ASME Section XI Inservice Inspection (ISI) detects many types of aging related to various aging degradation mechanisms). The "Generic Aging Lessons Learned" (GALL) report, which the Nuclear Regulatory Commission (NRC) plans to publish in the spring of 2001, evaluates existing plant programs generically to document the basis for determining when existing programs are adequate without change and when existing programs should be augmented for license renewal. Many of these programs were based on industry's consensus codes and standards. For example, the ISI program is based on

Section XI of the ASME Boiler and Pressure Vessel (B&PV) Code. However, in some cases, the Code may not have directly or explicitly addressed the aging management as required by 10 CFR Part 54, the license renewal rule. In some cases, there is no existing program to mitigate or prevent the aging effects expected during the period of extended operation. In these cases, the license renewal applicant develops a plant-specific aging management program and submits it to the NRC in its license renewal application for review and approval. This paper identifies areas where the ASME Code could be enhanced to provide a consistent approach in order to further streamline the license renewal review process.

Introduction and Background

The staff of the NRC has been developing three regulatory guidance documents for license renewal: NUREG-1801, "Generic Aging Lessons Learned (GALL) Report," NUREG-1800, "Standard Review Plan for License Renewal" (SRP-LR), and Regulatory Guide 1.188, "Standard Format and Content for Applications to Renew Nuclear Power Plant

Operating Licenses.” These documents are designed to streamline the license renewal review process by providing clear guidance for license renewal applicants and the NRC staff in preparing and reviewing license renewal applications. The GALL report systematically catalogs aging effects on structures and components, identifies the relevant existing plant programs, and evaluates the existing programs against the attributes considered necessary for an aging management program to be acceptable for license renewal. The GALL report also provides guidance for the augmentation of existing plant programs, including those based on the ASME Code, for license renewal. The revised SRP-LR allows an applicant to reference the GALL report to preclude further NRC staff evaluation if the plant’s existing programs are bounded by the aging management programs described in the GALL report. During the review process, the NRC staff will focus primarily on existing programs that should be augmented and new programs that should be developed specifically for license renewal. The GALL report identifies areas where existing programs, including codes and standards, could be enhanced to meet the needs of license renewal.

The objective of this paper is to give examples where the GALL report recommends that the ISI activities provided by ASME Section XI be enhanced for license renewal. Section 50.55a of title 10 of the Code of Federal Regulations (10 CFR 50.55a) endorses by reference the ISI in Section XI of the ASME B&PV Code. The ISI requirements have been shown to be generally effective in managing aging effects. In some cases, the ISI may not by itself be an effective aging management program for the license renewal period. For example, for managing the effects of crack initiation and growth due to stress corrosion cracking (SCC), the ISI program is used along with a water chemistry program. In other cases, the ISI needs to be enhanced to be an effective aging management program. For example, the VT-3 examinations of vessel internals have not identified bolt cracking because cracking occurs at the juncture of the bolt head and shank, which is not accessible for visual inspection. For some components, the VT-3 examination needs to be enhanced to detect aging in crevices and other inaccessible regions, or to detect cracks that are not obvious to the naked eye (tight or

hairline cracks). The GALL report identifies such circumstances in which the ASME Section XI ISI program, either by itself or in combination with another program, may not be an effective aging management program for license renewal and recommends augmentation of the existing program. The table below provides examples where ISI could be enhanced to give the license renewal applicants an aging management program that could be credited for license renewal.

The ISI program detects degradation of components by using the examination and inspection requirements specified in ASME tables. The program uses three types of examination: visual, surface, and volumetric. VT-1 examination detects discontinuities and imperfections, such as cracks, corrosion, wear, or erosion, on the surface of components. VT-2 examination detects evidence of leakage from pressure-retaining components during the system pressure test. VT-3 examination determines the general mechanical and structural condition of components and their supports by verifying parameters such as clearances, settings, and physical displacements, and detects discontinuities and imperfections, such as loss of integrity at bolted or welded connections, loose or missing parts, debris, corrosion, wear, or erosion. VT-3 includes examination for conditions that could affect the operability or functional adequacy of constant load and spring-type components and supports. Surface examination uses magnetic particle or liquid penetrant examinations to indicate the presence of surface discontinuities and flaws. Volumetric examination, such as radiography, ultrasonics, or eddy current, indicates the presence of discontinuities or flaws throughout the inside region of material included in the inspection program. In many cases, ISI is being credited in GALL as an existing program to detect aging effects. The extent and schedule of the inspection and test techniques prescribed by the program are designed to ensure structural integrity by discovering and repairing aging effects before the loss of intended function of the component. ISI can reveal crack initiation and growth, loss of material from corrosion, leakage of coolant, and indications of degradation due to wear or stress relaxation (such as clearances, setting, physical displacements, loose or missing parts, debris, wear, erosion, or the loss of integrity at

bolted or welded connections). The ASME code, Sections III and XI, have served the nuclear industry well in maintaining safety for existing plants. The NRC has been endorsing them by reference through 10CFR50.55a. However, compliance with the current rules in the consensus codes may not necessarily meet the requirements for an aging management program as delineated in the License Renewal Rule, 10CFR Part 54. For this reason, the staff has undertaken the GALL effort to identify where the consensus codes may be enhanced to be fully credited as an acceptable aging management program to support license renewal.

POTENTIAL ENHANCEMENTS

The following table lists various situations where components may experience age-related degradation and provides the recommendations from the GALL report for enhancing the ASME Code to manage the identified aging effects.

Table. Age-Related Degradation with GALL Recommendation and Basis

<p>AGE-RELATED DEGRADATION: Environmental Fatigue of reactor coolant pressure boundary components</p>
<p>GALL RECOMMENDATION: The fatigue design criteria for nuclear power plant components have changed as the industry consensus codes and standards evolved. The fatigue design criteria for a specific component depend on the version of the design code that applied to that component, that is, the code of record. There is a concern that the effects of the reactor coolant environment on the fatigue life of components were not adequately addressed by the code of record. Furthermore, the calculations supporting resolution of this issue and the nature of age-related degradation indicate that pipe leaks will become more frequent as plants continue to operate. Therefore, the GALL report recommends that an applicant address the effects of the coolant environment on component fatigue life in developing aging management programs to support license renewal. An applicant's consideration of the effects of coolant environment on component fatigue life for license renewal is currently an area of review.</p>
<p>AGE-RELATED DEGRADATION: Loss of fracture toughness due to neutron irradiation embrittlement in PWR reactor vessel internals and the vessel beltline shell and welds</p>
<p>GALL RECOMMENDATION: Loss of fracture toughness due to neutron irradiation embrittlement could occur in PWR reactor vessel internals and the vessel beltline shell and welds. Loss of fracture toughness is a consequence only if cracks exist. Since cracking is expected to initiate at the surface, the ASME Section XI ISI relies on VT-3 examination to detect cracks. However, VT-3 examination needs to be enhanced for creviced regions or for detecting tight cracks, and enhanced inspection and supplementary ultrasonic testing (UT) or other nondestructive examinations may be needed to effectively detect cracks. The GALL report recommends enhanced ISI to detect tight cracks and supplemental examinations for creviced regions. Stainless steels are susceptible to embrittlement under neutron irradiation. Fracture toughness depends strongly on the fluence on a particular component. Components may be screened out if the maximum tensile loading on the component under the ASME Code Level A, B, C, and D conditions are sufficiently low.</p>

AGE-RELATED DEGRADATION: Loss of fracture toughness due to thermal aging embrittlement of cast austenitic stainless steel (CASS) components

GALL RECOMMENDATION: The reactor coolant system components are inspected in accordance with the ASME B&PV Code, Section XI, Subsection IWB. This inspection needs to be enhanced to detect the effects of loss of fracture toughness due to thermal aging embrittlement of CASS components. An acceptable aging management program (AMP) consists of the following: determination of the susceptibility of CASS components to thermal aging embrittlement based on casting method, molybdenum content, and percent ferrite. For “potentially susceptible” components, aging management is accomplished either through enhanced volumetric examination or plant- or component-specific flaw tolerance evaluation. Additional inspection or evaluations to demonstrate that the material has adequate fracture toughness are not required for components that are not susceptible to thermal aging embrittlement.

AGE-RELATED DEGRADATION: Loss of fracture toughness due to thermal aging and neutron irradiation embrittlement of CASS reactor vessel internals

GALL RECOMMENDATION: The reactor vessel internals receive a visual inspection in accordance with Category B-N-3 of Subsection IWB, ASME Code Section XI. This inspection needs to be enhanced to detect the effects of loss of fracture toughness due to thermal aging and neutron irradiation embrittlement of CASS reactor vessel internals. In an acceptable AMP the applicant determines the susceptibility of CASS components to thermal aging embrittlement based on casting method, molybdenum content, and percent ferrite. For each “potentially susceptible” component, the applicant performs, as part of its 10-year ISI program during the license renewal term, a supplemental examination of the affected component based on the neutron fluence to which the component has been exposed or does a component-specific evaluation to determine the components susceptibility to loss of fracture toughness.

AGE-RELATED DEGRADATION: Crack initiation and growth due to SCC in pressurized water reactor (PWR) coolant system piping

GALL RECOMMENDATION: The GALL report recommends a plant-specific AMP for piping that does not meet either the reactor water chemistry guidelines of TR-105714 or the material guidelines of NUREG-0313. Crack initiation and growth from SCC could be significant for piping that does not meet the water chemistry or material guidelines and a plant-specific aging management program must be evaluated.

AGE-RELATED DEGRADATION: Boiling water reactor (BWR) internals cracking

GALL RECOMMENDATION: GALL recommends crediting the BWR vessel internals project (BWRVIP). BWRVIP-03 discusses the nondestructive examination (NDE) techniques for inspecting BWR vessel internals, their application, and the uncertainties of an NDE in a BWR. Crack initiation and growth due to SCC or intergranular stress corrosion cracking could occur in BWR shroud support structures. The inspection and flaw evaluation guidelines of BWRVIP-38 for shroud support are described in the staff-approved topical report. The GALL report recommends further evaluation of the BWRVIP-38 plant-specific program. The BWRVIP reports which were developed by industry include enhanced inservice inspection. The current version of the ASME Code could be enhanced to address the BWR internals cracking issue.

AGE-RELATED DEGRADATION: Crack initiation and growth due to SCC or irradiation-assisted stress corrosion cracking (IASCC) in baffle/former bolts

GALL RECOMMENDATION: Since only the heads of the baffle/former bolts are visible, the ASME Section XI VT-3 examination needs to be enhanced to detect relevant conditions of stress relaxation. Recent ultrasonic examinations of the baffle/former bolts have identified cracking in several plants. The GALL report recommends further evaluation to ensure these aging effects are adequately managed. Cracking has occurred in stainless steel (SS) baffle/former bolts in a number of foreign plants (IN 98-11) and has now been observed in U.S. plants.

AGE-RELATED DEGRADATION: Crack initiation and growth due to SCC or IASCC in PWR reactor vessel internals

GALL RECOMMENDATION: The existing program relies on ASME Section XI ISI to detect cracks and on control of water chemistry to mitigate SCC or IASCC. However, VT-3 examination may not be adequate to detect creviced regions or tight cracks, and the GALL report recommends that supplemental techniques be used to ensure that the component's intended function will be maintained during the extended period. As an alternative to enhanced inspection, the applicant may perform a component-specific evaluation, including a mechanical loading assessment, to determine the maximum tensile loading on the component during ASME Code Level A, B, C, and D conditions. Although SS components in PWRs have generally not been found to be affected by SCC because of low dissolved oxygen levels and control of primary water chemistry, there is potential for SCC from inadvertent introduction of contaminants into the primary coolant system (IN 84-18), from the introduction of relatively high levels of dissolved oxygen during shutdown, or from aggressive chemistries that may develop in crevice regions. Cracking has occurred in SS baffle/former bolts in a number of foreign plants (IN 98-11) and has been observed in U.S. plants.

AGE-RELATED DEGRADATION: Crack initiation and growth due to SCC and PWSCC in PWR core support pads, reactor vessel penetrations, and steam generator instrument and drain nozzles (Alloy 600)

GALL RECOMMENDATION: The GALL report recommends further evaluation to ensure these aging effects are adequately managed. The applicant is asked to provide a plant-specific program. PWR systems are susceptible to SCC in the presence of various corrodents. The reactor coolant system (RCS) of a PWR has a hydrogen overpressure maintained as an oxygen getter during power operation. As a result, the primary pressure boundary piping of PWRs has generally not been found to be affected by SCC. However, there are conditions with significant potential for inadvertent introduction of contaminants into PWR reactor coolant system. Ingress of demineralizer resins into the primary system has caused IGSCC of Alloy 600 vessel head penetrations (NRC IN 96-11, NRC GL 97-01). Inadvertent introduction of sodium thiosulfate into the primary system has caused IGSCC of steam generator tubes. The SCC has occurred in safety injection lines (NRC IN 97-19 and 84-18), charging pump casing cladding (NRC IN 80-38 and 94-63), instrument nozzles in safety injection tanks (NRC IN 91-05), and safety-related SS piping systems that contain oxygenated, stagnant, or essentially stagnant borated coolant (NRC IN 97-19). Steam generator tubes and plugs and Alloy 600 penetrations have experienced primary water stress corrosion cracking (PWSCC) (NRC IN 89-33, 94-87, 97-88, 90-10, and 96-11; NRC Bulletin 89-01 and its two supplements).

AGE-RELATED DEGRADATION: Crack initiation and growth due to thermal and mechanical loading or SCC in small-bore reactor coolant system and connected system piping

GALL RECOMMENDATION: The existing program relies on ASME Section XI ISI to detect cracks and on control of water chemistry to mitigate SCC. However, the ASME Section XI ISI does not require volumetric examination of pipes of less than 4-inch nominal diameter. The GALL report recommends a plant-specific destructive examination or a nondestructive examination that permits inspection of the inside surfaces of the piping to ensure that cracking has not occurred and the component's intended function will be maintained during the extended period. The AMP needs to be augmented by verifying that service-induced weld cracking is not occurring in the small-bore piping less than 4 inches in diameter, including pipe, fittings, and branch connections. Thermal and mechanical loading has led to cracking in high-pressure coolant injection piping (IN 89-80) and instrument lines (LER 50-249/99-003-2).

AGE-RELATED DEGRADATION: Changes in dimension due to void swelling in reactor internal components

GALL RECOMMENDATION: Changes in dimension due to void swelling could occur in reactor internal components. The GALL report recommends further evaluation to ensure this aging effect is adequately managed. The reactor vessel internals receive a visual inspection (VT-3). This inspection needs to be enhanced to detect the effects of changes in dimension due to void swelling. An acceptable AMP consists of participation in industry programs to address the significance of changes in dimensions due to void swelling and implementation of an inspection program if the results of the industry programs indicate the need for such inspections. The applicant should either provide the basis for concluding that void swelling is not an issue for the component or provide an AMP to manage the effects of changes in dimension due to void swelling and to the loss of ductility associated with swelling.

AGE-RELATED DEGRADATION: Loss of material due to pitting and crevice corrosion and crack initiation and growth due to thermal and mechanical loading or SCC in BWR isolation condenser components

GALL RECOMMENDATION: Loss of material due to pitting and crevice corrosion and crack initiation and growth due to thermal and mechanical loading or SCC could occur in BWR isolation condenser components. The existing program relies on control of reactor water chemistry to mitigate corrosion and on ISI to detect leakage. However, the existing program needs to be enhanced to detect cracking due to SCC and cyclic loading or loss of material due to pitting and crevice corrosion. The GALL report recommends augmenting the inspection program to include temperature and radioactivity monitoring of the shell side water and eddy current testing of isolation condenser tubes to ensure that the components' intended function will be maintained during the period of extended operation. Thermal fatigue and transgranular SCC have caused the isolation condenser tube bundles to fail (LER 50-219/98-014).

AGE-RELATED DEGRADATION: Loss of material due to pitting and crevice corrosion in the steam generator shell assembly

GALL RECOMMENDATION: The scope and schedule of the existing steam generator inspections are designed to ensure that flaws cannot attain a depth sufficient to threaten the integrity of the welds. However, based on NRC Information Notice 90-04, "Cracking of the Upper shell to Transition Zone Girth welds in Steam Generators," the program may not adequately detect pitting and corrosion. The GALL report recommends augmented inspection to manage this aging effect. Although the PWR primary pressure boundary has generally not been found to be affected by SCC because of low dissolved oxygen levels, there is a potential for SCC from inadvertent introduction of contaminants into the primary system (IN 84-18). Furthermore, IN 90-04 states: "If general corrosion pitting of the SG shell is known to exist, the requirements of Section XI of the ASME Code may not be sufficient to differentiate isolated cracks from inherent geometric conditions." Pitting has been reported at the PWR steam generator girth welds (NUREG/CR-4868). ASME Section XI requires only volumetric inspections of the girth welds to detect cracks. Additional examinations, i.e., visual and surface examinations, are recommended to detect pitting and general corrosion.

AGE-RELATED DEGRADATION: Cracking due to cyclic loading and SCC in containment penetrations

GALL RECOMMENDATION: Cracking of containment penetrations (including penetration sleeves, penetration bellows, and dissimilar metal welds) due to cyclic loading could occur in all types of PWR and BWR containments. A similar type of cracking could also occur in vent headers and downcomers due to SCC of BWR containments. These cracks are inspected by a VT-3 examination. However, this inspection may not detect fine cracks. The GALL report recommends further evaluation of programs to manage these aging effects.

AGE-RELATED DEGRADATION: Aging of inaccessible concrete and steel areas

GALL RECOMMENDATION: Increased porosity and permeability, cracking, and spalling due to leaching of calcium hydroxide and aggressive chemical attack, and cracking, spalling, loss of bond, and loss of material due to corrosion of embedded steel could occur in inaccessible areas of PWR concrete and steel containments, BWR Mark II concrete containments, and Mark III concrete and steel containments. The GALL report recommends further evaluation to manage the aging effects for inaccessible areas, when conditions do not exist in accessible areas that could indicate the presence of, or result in, degradation to such inaccessible areas.

The table above identifies a few examples from the GALL report, particularly pertinent to the ASME code. The NRC and industry are addressing these issues for license renewal. One way that individual license renewal applicants are addressing these issues are by developing plant specific programs that have to be reviewed and approved by the NRC. In some cases, applicants are expanding the scope of the ASME ISI program to address the issue. In other cases, the NRC is providing regulatory guidance to address the issue. A more desirable outcome is to enhance the code or standard to address the issue generically that could be endorsed by the NRC. Examples of

issues that may be addressed by ASME on a generic basis include:

- Crack initiation and growth in small bore reactor coolant and connected system piping
- Environmental fatigue of reactor coolant pressure boundary components

Other license renewal issues could also be addressed generically by ASME. This could promote consistency and streamline the license renewal process for both the industry and the NRC.

CONCLUSION

The ASME Code is a consensus document that has been widely used over many years. It has proven to be generally effective in detecting aging effects. However, as discussed above, the ASME Section XI ISI program, either by itself or in combination with another existing program, may require augmentation to be fully credited as an effective aging management program during the license renewal period. Addressing these issues through codes and standards could reduce the regulatory burden on license renewal applicants by providing a standard approach to resolve these issues and would further streamline the license renewal process for both the industry and the regulators.

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