February 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 paper "POTENTIAL ENHANCEMENTS OF CODES AND STANDARDS TO SUPPORT LICENSE RENEWAL" for Track 10, "NRC/ASME Symposium on BPVC Section XI," for your review. 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

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 paper "POTENTIAL ENHANCEMENTS OF CODES AND STANDARDS TO SUPPORT LICENSE RENEWAL" for Track 10, "NRC/ASME Symposium on BPVC Section XI," for your review. 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

DISTRIBUTION:
See attached page

ML010390348

DOCUMENT NAME: G:\RLSB\DOZIER\ICONELETTER.WPD

OFFICE	LA:DRIP	GE:RLSB:DRIP	SC:RLSB:DRIP	SC:RLSB:DRIP
NAME	EGHylton	IJDozier	SSLee	PTKuo
DATE	01/ 8/01	01/ 9/01	01/9/01	02/2/01
OFFICE	C:RLSB:DRIP	D:DRIP		
NAME	CIGrimes	DBMatthews		
DATE	02/2/01	02/5/01		

OFFICIAL RECORD COPY

DISTRIBUTION: **ML010390348**

HARD COPY

RLSB RF
E. Hylton
J. Dozier

E-MAIL:

PUBLIC (ADAMS)
D. Matthews
S. Newberry
C. Grimes
C. Carpenter
J. Strosnider
R. Wessman
G. Bagchi
K. Manoly
W. Bateman
M. Mayfield
S. Bahadur
RLSB Staff

J. Vora

Submitted to ASME PVP 2001 Conference for Publication in August, 2001. This Preprint (draft) is not to be cited or reproduced. This is a preprint (draft) of a paper intended for publication in a journal or proceedings. Since changes may be made before publication, this preprint (draft) is made available with the understanding that it will not be cited or reproduced without the permission of the author.

POTENTIAL ENHANCEMENTS OF CODES AND STANDARDS TO SUPPORT LICENSE RENEWAL

Jerry Dozier, Sam Lee, P. T. Kuo
U.S. Nuclear Regulatory Commission

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). Also, since ASME Section XI inspection typically requires volumetric or surface examination of only the welds or weld regions, the potential for cracking in regions remote from the welds is sometimes not addressed. 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, liquid penetrant, or eddy current 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 following table lists various situations where components may experience age-related degradation and gives 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

AGE-RELATED DEGRADATION: Environmental Fatigue
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.
AGE-RELATED DEGRADATION: Loss of fracture toughness due to neutron irradiation embrittlement 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 loss of fracture toughness. The GALL report recommends further evaluation to ensure that this aging effect is adequately managed. Loss of fracture toughness can occur because of the high fluence level experienced by the baffle/former bolts.
AGE-RELATED DEGRADATION: Loss of fracture toughness due to neutron irradiation embrittlement in PWR reactor vessel internals and the vessel beltline shell and welds
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.
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 determinates 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) CASS or 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 irradiationassisted 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 of the safety injection tank, refueling water tank, and associated components in PWRs

GALL RECOMMENDATION: The existing plant program relies on inservice visual inspection and water chemistry to detect and mitigate degradation. However, visual inspection cannot detect cracks initiated on the inside surface. Therefore, verification of the effectiveness of the ISI and chemistry control programs should be performed to ensure that corrosion is not occurring. The GALL report recommends further evaluation of programs to manage crack initiation and growth due to SCC to verify the effectiveness of the ISI and chemistry control programs. A one-time inspection of select components and susceptible locations is an acceptable method to ensure that corrosion is not occurring and the component's intended function will be maintained during the period of extended operation.

AGE-RELATED DEGRADATION: Crack initiation and growth due to primary water stress corrosion cracking (PWSCC) in PWR pressurizer instrumentation penetrations and heater sheaths and sleeves made of Ni-alloys (Alloy 600)

GALL RECOMMENDATION: The existing program relies on ASME Section XI ISI to detect cracks and on control of water chemistry to mitigate PWSCC. However, the combination of these two programs does not fully manage the aging effects of SCC on the intended function of Ni-alloy components. The GALL report recommends that the applicant evaluate the susceptibility of Ni-alloys to PWSCC and perform a susceptibility study of all Ni-alloy components to identify the most susceptible locations and to determine whether an augmented inspection program, including a combination of surface and volumetric examination, is necessary. The applicant should review its leakage detection system and the scope and schedule of its inspection program to ensure that cracks are detected before the loss of the intended function of the penetrations. The GALL report recommends that the applicant either provide a technical basis to justify the adequacy of the program or develop an integrated long-term program to periodically inspect the locations most susceptible to PWSCC. SCC of Alloy 600 and austenitic SS has occurred in domestic and foreign PWRs (IN 90-10). PWSCC of Alloy 600 is not a new phenomenon.

AGE-RELATED DEGRADATION: Crack initiation and growth due to SCC and PWSCC in PWR core support pads, reactor vessel penetrations, pressurizer spray heads, the flange leak detection line, and steam generator instrument and drain nozzles.

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 two conditions with significant potential for inadvertent introduction of contaminants into PWR reactor coolant system. The first condition is an unacceptable level of contaminants in the boric acid purchased. The second is the free surface of the spent fuel pool, which can be a natural collector of airborne contaminants. During refueling operations there is direct communication between the reactor coolant system and the spent fuel pool, and this is more free surface to collect any airborne contaminants from concurrent maintenance activities.

Other systems that utilize borated water and are made with austenitic materials may not receive the same attention as the RCS. These systems are extensively cross-connected, and some equipment has more than one system function. Thus, contaminants introduced at one point may appear elsewhere. Because of inadvertent safety injection actuation, potentially contaminated water can enter the reactor coolant system. In September 1980, the NRC published NUREG-0691, "Investigation and Evaluation of Cracking Incidents in Piping in Pressurized Water Reactors." The NUREG discusses pipe cracking from a variety of causes in austenitic and nonaustenitic materials and gives information on cracks discovered through May 1980. Additional information is given in NUREG-0679, published in August 1980. Since the publication of NUREG-0691 and NUREG-679, additional instances of stress-corrosion attack have been reported.

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 preload due to stress relaxation in PWR reactor vessel internal bolts and screws

GALL RECOMMENDATION: The ASME Section XI inspection relies on VT-3 examination to reveal indications of degradation due to stress relaxation such as loose or missing parts, wear, or debris. However, VT-3 examination may not be adequate to detect loss of mechanical closure integrity and enhanced inspection techniques and augmented inspection programs are needed to ensure that the component's intended function will be maintained during the period of extended operation. The GALL report recommends enhanced ISI to detect loss of mechanical closure integrity and an augmented inspection program to determine critical locations and monitoring techniques.

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
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 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.

These examples are from the August 2000 draft version that was issued for public comment. These GALL recommendations are subject to change based on the evaluation of public comments. GALL is scheduled to be officially issued during the Spring of 2001.

The NRC staff believes that by addressing the above recommendations, the ASME Code could be pro-active and responsive to industry needs and could substantially increase the operating life of passive components in nuclear power plants.

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 ensure the detection of aging effects in certain cases before the loss of the intended function of the component during the period of license renewal. Currently, the license renewal applicant must propose plant-specific programs to address these issues. 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.

References

ASME Section XI, "*Rules for Inservice Inspection of Nuclear Power Plant Components*," the ASME Boiler and Pressure Vessel Code, 1989 edition, the American Society of Mechanical Engineers, New York, NY.

ASME Section XI, "*Rules for Inservice Inspection of Nuclear Power Plant Components*," the ASME Boiler and Pressure Vessel Code, 1995 edition, the American Society of Mechanical Engineers, New York, NY.

BWRVIP-03, "*BWR Vessel and Internals Project, Reactor Pressure Vessel and Internals Examination Guidelines*, (EPRI TR-105696 R1)" BWRVIP and Electric Power Research Institute, Palo Alto, CA, March 30, 1999.

BWRVIP-62, "*BWR Vessel and Internals Project, Technical Basis for Inspection Relief for BWR Internal Components with Hydrogen Injection*" (EPRI TR-108705), BWRVIP and Electric Power Research Institute, Palo Alto, CA, December 31, 1998.

EPRI TR-105714, *PWR primary Water Chemistry Guidelines-Revision 3*, Electric Power Research Institute, Palo Alto, CA, Nov. 1995.

IE Bulletin No. 80-13, *"Cracking in Core Spray Spargers*," May 12, 1980.

LER 50-249/99-003-1, "Pressure flow element venturi instrument line steam leakage resulting in Unit 3 shutdown, due to fatigue of socket welded pipe joint,"

LER 50-219/98-014, "Failure of isolation condenser tube bundles caused by thermal stresses/TGSCC,"

NUREG-1800, "Standard Review Plan (SRP) for the Review of License Renewal Applications for Nuclear Power Plants," Working Draft, NRC August 2000 Draft Report.

NUREG-1801, "Generic Aging Lessons Learned" (GALL), NRC August 2000 Draft Report , (available on the internet at http://www.nrc.gov/NRC/REACTOR/LR/index.html).

NEI 95-10, Revision 2, *Industry Guideline for Implementing the Requirements of 10 CFR Part 54 – The License Renewal Rule*, Nuclear Energy Institute, August 2000 Draft Report.

NRC Information Notice 80-38, *"Cracking in Charging Pump Casing Cladding*," October 31, 1980.

NRC Information Notice 82-37, *“Cracking in the Upper Shell to Transition Cone Girth Weld of a Steam Generator at an Operating PWR*,” September 16, 1982.

NRC Information Notice 84-18, *“Stress Corrosion Cracking in PWR Systems*,” March 7, 1984.

NRC Information Notice 85-65, *“Crack Growth in Steam Generator Girth Welds*,” July 31, 1985.

NRC Information Notice 89-80, *“Potential for Water Hammer, Thermal Stratification, and Steam Binding in High-Pressure Coolant Injection Piping*,” December 1, 1989.

NRC Information Notice 90-04, *“Cracking of the Upper-Shell-to-Transition Cone Girth Welds in Steam Generators*,” January 26, 1990.

NRC Information Notice 91-05, *“Intergranular Stress Corrosion Cracking in Pressurized Water Reactor Safety Injection Accumulator Nozzles*,” January 30, 1991.

NRC Information Notice 94-63, *“Boric Acid Corrosion of Charging Pump Casing Caused by Cladding Cracks*,” August 30, 1994.

NRC Information Notice 95-17, *“Reactor Vessel Top Guide And Core Plate Cracking*,” March 10, 1995.

NRC Information Notice 97-19, *“Safety Injection System Weld Flaw at Sequoyah Nuclear Power Plant, Unit 2*,” April 18, 1997.

NRC Information Notice 97-46, *“Unisolable Crack in High-Pressure Injection Piping*,” July 9, 1997.

NRC Information Notice 97-88, *“Experiences During Recent Steam Generator Inspections*,” December 16, 1997.

NRC Information Notice 98-11, *“Cracking of Reactor Vessel Internal Baffle Former Bolts in Foreign Plants*,” March 25, 1998.

NRC Inspection Report 50-255/99012, “Palisades Nuclear Generating Plant NRC Inspection Report,”

NRC Generic Letter 94-03, *“Intergranular Stress Corrosion Cracking of Core Shrouds in Boiling Water Reactors*,” July 25, 1994.

Regulatory Guide RG-1.188,”Standard Format and Content for Applications to Renew Nuclear Power Plant Operating Licenses,” NRC August 2000 Draft Report.

SECY-99-148, "Credit For Existing Programs For License Renewal," June 3, 1999.

10 CFR 50.55a, *“Codes and Standards*,” Office of the Federal Register, National Archives and Records Administration, 2000.