Aging Management of Reactor Coolant System Mechanical Components for License Renewal[1]

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Abstract

The reactor coolant system (RCS) mechanical components that require an aging management review for license renewal include the primary loop piping and associated connections to other support systems, reactor vessel, reactor vessel internals, pressurizer, steam generators, reactor coolant pumps, and all other inter-connected piping, pipe fittings, valves, and bolting. All major RCS components are located inside the reactor building. Based on the evaluation findings of recently submitted license renewal applications for pressurized water reactors, this paper presents the plant programs and/or activities proposed by the applicants to manage the effects of aging. These programs and/or activities provide reasonable assurance that the intended function(s) of these mechanical components will be maintained for the period of extended operation. The license renewal application includes identification of RCS subcomponents that are within the scope of license renewal and are vulnerable to age-related degradation when exposed to environmental and operational conditions, determination of the effects of aging on their intended safety functions, and implementation of the aging management programs and/or activities including both current and new programs. Industry-wide operating experience, including generic communication by the NRC, is part of the aging management review for the RCS components. In addition, this paper discusses time-limited aging analyses associated with neutron embrittlement of the reactor vessel beltline region and thermal fatigue.

Introduction

The License Renewal Rule, 10 CFR Part 54[1], contains the regulatory requirements that must be satisfied to obtain a renewed operating license which allows continued operation of a nuclear power plant beyond its original license term. The Rule is founded on two principles[2]: (1) the regulatory process is adequate to ensure that the licensing bases of all currently operating plants provide and maintain an acceptable level of safety so that operation will not be inimical to public health and safety or common defense and security, and (2) the plant-specific licensing basis must be maintained during the renewal term in the same manner and to the same extent as during the original licensing term. In addition to the identification and evaluation of time-limited aging analyses (TLAAs), the focus of the Rule is on providing reasonable assurance that the effects of aging on the functionality of long-lived passive structures and components are adequately managed in accordance with the plant-specific design basis conditions.

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To determine the safety-related structures, systems, and components (SSCs) that are within the scope of the Rule, the applicant must identify those SSCs that are relied upon to remain functional during and following the plant-specific design basis events (DBEs), consistent with the current licensing basis (CLB) of the facility[3,4]. Once the SSCs within the scope of the license renewal are identified, the licensee determines the structures and components that are subject to an aging management review (AMR). The AMR for a structure or component is directly related to whether the structure or component performs an intended function without moving parts or without a change in configuration or properties (i.e., it is passive) and that is not subject to replacement based on a qualified life or specified time period (i.e., it is long-lived). The applicant must demonstrate that the effects of aging associated with these structures and components will be adequately managed during the period of extended operation.

In a pressurized water reactor (PWR), the reactor coolant system (RCS) mechanical components that require an AMR as part of the license renewal application (LRA) include: the primary loop piping and associated connections to other support systems, reactor vessel including the control rod drive components, reactor vessel internals, pressurizer including safety relief valves and pressure relief tank, steam generators, reactor coolant pumps, and all other inter-connected piping, pipe fittings, valves, and bolting. All major components of the RCS are located in the reactor building. During operation, the RCS transfers heat generated in the core to the steam generators where steam is produced to drive the turbine-generator. Borated demineralized water is circulated in the RCS at a flow rate and temperature consistent with achieving the reactor core thermal-hydraulic performance. The water also acts as a neutron moderator.

The RCS pressure boundary provides a barrier against the release of radioactivity generated within the reactor, and is designed to ensure a high degree of integrity throughout the life of the reactor unit. The RCS pressure is controlled by the use of the pressurizer where water and steam are maintained at saturated conditions. Pressure is raised or lowered by the use of electrical heaters or water sprays.

As of December 2001, eight LRAs have been submitted to the NRC for approval. Brookhaven National Laboratory (BNL) reviewed the RCS section of three of these LRAs. One LRA was submitted for a Babcock & Wilcox (B&W) designed plant[5], and the other two LRAs were submitted for two multi-plant sites with each site containing two Westinghouse (W) designed plants[6,7]. One out of the eight LRAs submitted for a Combustion Engineering (CE) plant has been reviewed and approved for extended license[8]. This paper presents insights obtained from the evaluation of these LRAs, including the applicable effects of aging and the aging management programs and/or activities for the RCS mechanical components[9].

Aging Management of RCS Mechanical Components

The B&W owners group (B&WOG) and W owners group (WOG) have described their aging management programs and/or activities that are applicable to the RCS components in a series of topical reports. These topical reports were previously reviewed and approved by the NRC staff. The staff determined that the topical reports presented adequate information to meet the requirements of 10 CFR 54.21(a)(3) for managing the effects of aging in RCS mechanical components. The final safety evaluation report (FSER) issued by the staff for each topical report contains a discussion of action items that should be addressed if an applicant decides to use these topical reports as part of its LRA. Thus, an applicant may incorporate these reports by reference if the conditions of approval in the FSER of the specific report are met. Although the owners group topical reports were available for all three LRAs reviewed by BNL, one W PWR applicant did not use them as part of its LRA.

In general, the applicant begins its AMR by identifying applicable aging effects for the structures and components (SCs) that are subject to an AMR by reviewing the potential aging mechanisms identified in the industry literature, generic communications by the NRC, vendors and manufacturers, and plant-specific operating experience. Component materials, operating environment, and operating stresses are used to determine the applicable aging effects for each subcomponent of the RCS major components within the scope of the license renewal. The applicant then reviews its existing aging management programs and/or activities for the SCs that are subject to an AMR, and identifies those programs that can be used to manage the applicable effects of aging. The applicant either demonstrates the effectiveness of the different programs and activities consistent with 10 CFR 54.21(a)(3), or develops new programs or activities to manage the remaining applicable effects of aging. The applicant provides descriptions of the aging management programs (AMPs) that will be used to manage the effects of aging in accordance with the ten attributes given in the NRC's standard review plan for license renewal (SRP-LR)[10].

The Class 1 piping, valves, pumps, and other RCS components that are subject to an AMR have been designed to meet the requirements of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code, Section III, Subsection NB for Class 1 components[11] or the American National Standards Institute (ANSI) B31.7 Code[12]. The predominant material of construction for the Class 1 components is stainless steel including cast austenitic stainless steel (CASS), carbon steel, low alloy steel, and alloy materials (e.g., Alloy 82/182, Alloy 600/690). The internal surfaces of all Class 1 components wetted by borated water are stainless steel. Some bolting and exterior surfaces of the pressure boundary components are identified as carbon or low alloy steel. Design and welding considerations in the selection of materials for RCS components reduce the susceptibility of Class 1 piping and component materials to sensitization.

The Class 1 components that are within the scope of license renewal are internally exposed to borated reactor coolant water at approximately 650 °F and 2250 psig. These components are located in the reactor building (i.e., containment) and are externally exposed to an air environment. External surfaces near mechanical piping connections (e.g., flanges) may also be exposed to borated water leakage condition.

Based on the requirements of 10 CFR 54.4(a), one of the intended functions for the Class 1 RCS components is to maintain mechanical pressure boundary integrity.

The RCS mechanical components are grouped into five specific major items that are within the scope of the license renewal. They include the RCS Class 1 piping and associated components including the reactor coolant pumps, the reactor vessel, the reactor vessel internals, the pressurizer, and the steam generators. For each of these mechanical components, this paper presents its long-lived passive subcomponents that require an AMR, the potential effects of aging associated with these subcomponents, and the aging management programs and/or activities that are proposed by the applicants in order to manage these effects of aging. This paper also provides a discussion of the applicable time-limited aging analyses (TLAAs) for these major RCS components. In addition, this paper highlights a number of generic issues associated with each RCS component that require further review by the staff.

RCS Class 1 Piping and Associated Components

Scope

The scope of the AMR for RCS Class 1 piping and associated components includes

- the Nuclear Steam Supply System

(NSSS)-supplied primary loop Class 1 piping of the RCS pressure boundary that are connected to the reactor vessel, the steam generators (primary side), and the reactor coolant pump;

- the Balance of Plant (BOP)-designed Class 1 piping of other support systems that are attached to the primary loop piping;
- pressure boundary portion of Class 1 valves (bodies and bonnets, bolting); and
- pressure boundary portion of the reactor coolant pump.

The NSSS-supplied primary piping includes branch connection nozzles and special items such as resistance temperature detector (RTD) scoop elements, pressurizer spray scoop, sample connection scoop, the reactor coolant temperature element installation boss, and the temperature element well itself. Class 1 branch piping consists of piping connected at the NSSS-supplied primary loop piping out to and including (1) the outermost containment isolation valve in the piping which penetrates primary containment, or (2) the second of two valves normally closed during normal reactor operation, in piping which does not penetrate primary containment. For Class 1 valves, the pressure-retaining portion of the component consists of the valve body, bonnet and closure bolting. For the reactor coolant pumps (RCPs), the pressure-retaining portion includes the pump casing, the main closure flange, the thermal barrier heat exchanger within the RCP, the RCP seals and the pressure retaining bolting.

Effects of Aging

The internal service environment for the Class 1 piping and associated components is borated water, while the external environment is reactor building atmospheric air. The aging effects applicable to the Class 1 piping and associated components that require aging management are:

- cracking and loss of material of stainless steel components (including CASS) in borated water environment.
- loss of material of carbon steel and low-alloy steel components in treated water environment.
- loss of material from carbon steel and low-alloy steel components in borated water leakage environment.
- reduction in fracture toughness of CASS components (including valve bodies and bonnets, and pumps) in a high-temperature borated water environment.
- loss of pre-load of stainless steel and low alloy steel bolting in the reactor building air environment.

AMPs and TLAAs

The applicable AMPs for the Class 1 piping and associated components are:

- Water Chemistry Control Program
- Boric Acid Corrosion Surveillance Program
- Inservice Inspection Program

The TLAA applicable to Class 1 piping and associated components is metal fatigue.

Highlights of Review Findings

In accordance with the ASME Section XI[13] inservice inspection (ISI) requirements for Class 1 components, pipe sizes 4 inch nominal pipe size (NPS) and above are subject to both volumetric and surface examinations in addition to leakage detection. Piping >1-inch and < 4-inch NPS is subject to only surface examination and leakage detection. Piping ≤1-inch NPS is subject to leakage detection only. Operating experience has shown that cracking originates from the inside surface of the piping and therefore, any through-wall crack in piping <4 inch NPS may be left unnoticed until the piping fails under a design basis loading. For a reasonable assurance that the cracking in small-bore piping is adequately managed, an augmented inspection program may be necessary.

Components fabricated from cast austenitic stainless steel (CASS) that have delta ferrite levels below the susceptibility screening criteria have adequate fracture toughness and do not require any supplemental inspection. As a result of thermal embrittlement, components that have delta ferrite levels exceeding the screening criterion (delineated in a May 19, 2000 letter to NEI from the NRC)[14] may not have adequate fracture toughness and may require additional evaluation or examination. Also, the current ISI activities are sufficient for RCP casing and valve bodies, but inadequate for the piping base metals (for each heat of material) which are not subject to an ISI. Therefore, an additional evaluation or examination is warranted. A detailed evaluation to demonstrate the safety and serviceability of these components fabricated from CASS may be necessary.

The RCP seals are typically excluded from an AMR because they are periodically replaced. Preventive maintenance is currently scheduled periodically for the RCP seals unless data indicates that the inspection must be done more frequently. The RCP seals can be excluded provided the applicant's program consists of performance and condition monitoring activities that provide reasonable assurance that the intended function of the RCP seals will be maintained in the period of extended operation. Also, the plant operating experience should demonstrate the effectiveness of these activities.

Metal fatigue is an applicable TLAA for all RCS components. A TLAA based on 10 CFR 54.21(c)(1), option (i) or (ii), would require the applicant to demonstrate that cumulative usage factors (CUF) remain less than 1.0 when the period of extended operation is considered. NUREG/CR-6260[15] identified several locations of RCS components where the CUF may exceed 1.0 when considering the environmental effects. The industry recently proposed an existing or enhanced ISI program to manage cracking due to fatigue for locations where the CUF of 1.0 may be exceeded. This includes procedures adequate to detect flaw sizes that can be shown not to propagate to failure between inspection intervals. The staff has not endorsed this position on a generic basis at this time.

In summary, the scope for the RCS piping and associated components may vary depending on the design configuration of the RCS in the plant. One plant has included the letdown heat exchangers within the scope and another plant included both ASME Class 1 and Class 2 components inside the reactor building. However, all three applications reviewed by BNL considered the same major RCS piping and associated components within the scope of license renewal and also, they all have identified the same aging effects associated with them. Typically, the Water Chemistry Control Program and the Inservice Inspection Program are identified to manage the loss of material and cracking in RCS piping components. For all components made out of carbon steel or low alloy steel, the loss of material due to boric acid leakage is managed by the Boric Acid Corrosion Surveillance Program.

Reactor Vessel

Scope

Each reactor vessel (RV) contains the core, core supporting structures, control rods and other parts directly associated with the core. The RV upper head contains a number of penetrations (a large number of them for control rod drive mechanism (CRDM) penetrations and some for auxiliary head adapters). Each RV has an inlet nozzle and an outlet nozzle for each of the primary piping loops located just below the flange. Coolant enters through the inlet nozzles, flows down the core barrel-vessel wall annulus, turns at the bottom and flows up through the core to the outlet nozzles. The bottom head has penetrations for connection and entry of in-core instrumentation.

Effects of Aging

The service environment for the RV and CRDM pressure boundary components is borated water. The environment for the RV integral attachments, the RV head closure studs and the external surfaces is the reactor building atmospheric condition. The aging effects applicable to the reactor vessel components that require aging management are:

- cracking of stainless steel (including CASS), carbon steel, low alloy steel, and nickel-based alloy steel in borated water environment.
- loss of material from carbon steel and low-alloy steel components in borated water leakage environment.
- reduction in fracture toughness of CASS components in a high-temperature borated water environment.
- loss of pre-load of stainless steel and low alloy steel bolting in reactor building air environment.

AMPs and TLAAs

The applicable AMPs for the RV and CRDM pressure boundary components are:

- Water Chemistry Control Program
- Inservice Inspection Program
- Alloy 600 Aging Management Program
- Reactor Vessel Integrity Program
- CRDM Nozzle and Other Vessel Closure Penetrations Inspection Program
- Reactor Coolant System Operational Leakage Monitoring Program
- Bottom Mounted Instrumentation Thimble Tube Inspection Program
- Boric Acid Corrosion Surveillance Program.

Two TLAAs are applicable to the reactor vessel. One TLAA addresses reactor vessel neutron embrittlement which includes upper shelf energy, pressurized thermal shock, and pressure-temperature limits. The other TLAA addresses metal fatigue.

Highlights of Review Findings

License renewal issue number 98-0012[16], "Consumables" was issued to the Nuclear Energy Institute (NEI) by letter dated April 20, 1999. In this position the staff stated that packing, gaskets, seals, and O-rings are not typically required by the CLB to fulfill the functions of 10 CFR 54.4(a)(1)(i). The position also stated that according to ASME, Section III, NB-2121, NC-2121, and ND-2121, packing, gaskets, seals, and O-rings are not relied upon for a pressure retaining function in components for which these Code design practices apply. Inasmuch as these Code design practices do not apply to the O-ring in the closure head, and since the sealing surface of the vessel flange is the pressure boundary, the O-rings are not within the scope of 10 CFR Part 54. However, because the monitoring pipes from the RPV closure head penetrate the sealing surfaces of the vessel flanges, they should be treated as part of the reactor coolant system pressure boundary and, therefore, are within the scope of 10 CFR Part 54. The industry response to this issue was that the monitoring pipes do not directly support the RCS pressure boundary and, therefore, are not within the scope of license renewal. Also, if the inner reactor vessel closure flange O-ring fails, and RCS fluid is introduced into the monitoring pipes, leak flow would be limited because the ½-inch diameter orifice holes in the flange are smaller than the inside diameters of these pipes. In addition, any leakage through the monitoring pipes, during normal operation is estimated by the applicant to be within the makeup system capacity of the RCS.

NRC Bulletin No. 88-09[17] and Information Notice No. 87-44[18] identified flow-induced vibration as a cause for wear (i.e., thinning) of the thimble tubes, resulting in degradation of the RCS pressure boundary and potentially leading to non-isolable leak of reactor coolant. The amount of vibration the thimble tubes experience is determined by plant-specific factors such as the gap distance from the lower core plate to the fuel assembly instrument tube, the amount of clearance between the thimble tube and the guide or instrument tube, the axial component of the local fluid velocity, the thickness of the thimble tube, and the moment of inertia of the thimble tube. The only effective method for determining thimble tube integrity is through plant-specific inspections and periodic monitoring.

Primary water stress corrosion cracking (PWSCC) in the axial direction of control rod drive mechanism (CRDM) nozzles has previously been observed but has not been considered a safety concern requiring immediate attention. This issue was addressed by the NRC in Generic Letter (GL) 97-01[19], issued in 1997. Inspections and repairs have generally been made during plant maintenance outages. However, in 2001 circumferential cracking was discovered in the CRDM base metal above the structural retaining weld at some PWRs. The flaw with the largest circumferential extent (165°) was found at Oconee Unit 3. Cracking of the CRDM nozzles represents a degradation of the primary RCS boundary, and hence, is potentially safety significant. On April 30, 2001 the staff issued Information Notice 2001-05[20] relating to the through-wall circumferential cracking of the RV head CRDM nozzles at Oconee 3. In addition, the NRC issued Bulletin 2001-01[21], "Circumferential Cracking of Reactor Pressure Vessel Head Penetration Nozzles," on August 3, 2001 due to the discoveries of cracked and leaking RPV nozzles that are fabricated from Alloy 600 material. The bulletin requested information from the licensees of 69 PWRs regarding the structural integrity of their RPV head penetrations (i.e. including current and future inspection plans) in order to determine if additional regulatory actions are required by the NRC.

The current aging management activities, including the chemistry control and existing ASME Section XI inspections (Examination Category B-N-2, VT-3 visual examination), are not adequate for managing cracking of the core support lugs. The industry is currently in the process of developing new recommendations for inspection of core support lugs. Applicants should stay abreast of the industry recommendations to determine the need for enhanced inspection of core support lugs.

In summary, the reactor vessel is the most important component within the RCS and its internal surfaces are exposed to both high neutron radiation and high temperature, specifically near the beltline region surrounding the core. Both top and bottom vessel heads contain many penetrations to insert control rod drive mechanisms through the top head and to insert thimble tubes through the bottom head. In addition, the vessel contains primary coolant nozzles and several other nozzles, including the core flooding nozzles. Each penetration through the vessel is a source for cracking at the vessel-nozzle interface. All applicants have appropriately identified the aging effects associated with these penetrations and the AMPs to adequately manage them. Another issue that is considered important for the reactor vessel is the loss of fracture toughness in the beltline region material due to both high neutron flux and high temperature conditions. The TLAA on neutron embrittlement and the AMP on reactor vessel integrity address this particular aging effect.

Reactor Vessel Internals

Scope

The reactor vessel internals (RVIs) consist of three parts: the lower core support structure, the upper core support structure, and the in-core instrumentation support structure. The lower core support structure consists of the core barrel; the core baffle; the lower core plate and support columns; the neutron shield pads; and the core support which is welded to the core barrel. The lower core support structure is supported at its upper flange by a ledge in the reactor vessel, and its lower end is restrained from transverse motion by a radial support system attached to the vessel wall. The upper core support structure, which is removed as a unit during refueling, consists of the upper support assembly and the upper core plate. The upper head injection (UHI) support columns and guide tube assemblies are between the upper support assembly and the upper core plate. The upper core support assembly is positioned in its proper orientation with respect to the lower support structure by slots in the upper core plate which engage the upper core plate alignment pins. The in-core instrumentation support structures consist of an upper system to convey and support thermocouples penetrating the vessel through the head and a lower system to convey and support flux thimbles penetrating the vessel through the bottom.

The RVIs support the core, maintain fuel alignment, limit fuel assembly movement, maintain alignment between fuel assemblies and control rod drive mechanisms, direct coolant flow past the fuel elements, direct coolant flow to the pressure vessel head, provide gamma and neutron shielding, and provide guides for the in-core instrumentation.

Effects of Aging

The service environment for the RVI components is borated water. The aging effects applicable to the reactor vessel internal components that require aging management are:

- cracking of stainless steel (including CASS) and nickel-based alloy components in a borated water environment.
- loss of material from stainless steel (including CASS) and nickel-based alloy components in a borated water environment.
- reduction in fracture toughness of stainless steel (including CASS) in a high temperature and high radiation borated water environment.
- loss of preload of stainless steel bolting and hold down springs in a borated water environment.
- dimensional changes of stainless steel components in a borated water environment.

AMPs and TLAAs

The applicable AMPs for the RVI components are:

- Water Chemistry Control Program
- Inservice Inspection Program
- Reactor Vessel Internals Inspection Program
- Alloy 600 Aging Management Program

The TLAA applicable to RVI components is metal fatigue.

Highlights of Review Findings

Void swelling is defined as a gradual increase in dimensions of the RVI components. Under reactor internals irradiation conditions, helium is generated through nuclear transmutation reactions. At sufficiently high temperatures, helium bubbles expand to a critical diameter, whereby the helium bubbles convert to bias-driven voids. Void formation results in the swelling of the material. A change in dimensions may affect the ability of the RVI components to perform their intended functions. Although void swelling has not been observed to date, void swelling may become significant during the period of extended operation. Until the industry has developed sufficient data to demonstrate that void swelling is not a significant aging mechanism, void swelling should be considered significant, and applicants for license renewal should describe their aging management plan to address void swelling.

Inservice inspections are designed to manage cracking of welded joints, cracking of bolting material, and general loss of material. Examination Category B-N-3, for removable core support structures, is directly applicable to the RVI components. The examination requirement is a visual VT-3 of all accessible parts of the RVI components. Cracks initiated by SSC or fatigue are expected to start off very small and then grow over time. VT-3 examination may not be adequate for detecting cracks before they reach the critical flaw size. An augmented examination for these components may be necessary.

The Reactor Vessel Internals Inspection Program manages cracking due to irradiation-assisted stress corrosion cracking (IASCC) and SCC, reduction in fracture toughness due to irradiation and thermal embrittlement, dimensional changes due to void swelling, and loss of preload due to stress relaxation. The scope of this AMP consists of three groups of stainless steel RVI components: (1) plates, forgings, and welds, (2) bolting (baffle-to-baffle, baffle-to-former, and barrel-to-former), and (3) CASS components. The AMP will incorporate insights gained from industry activities through the Electric Power Research Institute (EPRI) and other industry groups. This is a new program to be developed for the period of extended operation.

Inspections monitor crack propagation of existing flaws caused by reduction in fracture toughness due to thermal embrittlement and irradiation embrittlement; for items fabricated from CASS. It is not clear how applicants will address reduction in fracture toughness of CASS components due to the combined effects of thermal aging embrittlement and irradiation embrittlement. A valid analysis to predict critical crack size requires knowledge of the expected fracture toughness as a function of exposure time to both temperature and irradiation. With increasing time of operation, the critical crack size will tend to decrease, as the fracture toughness decreases. An end-of-extended-license prediction would appear to be the most reasonable approach. In this case, the critical crack size may be undetectable by visual examination.

In summary, the RVI components are totally immersed in borated primary water inside the reactor vessel and are exposed to high radiation and high temperature conditions, specifically near the core region. All components are fabricated from stainless steel and there is limited access to the RVI subcomponents for inspection. The present inservice inspection (ISI) requirements are considered to be inadequate and an augmented inspection may be necessary to fully understand the degradation in a number of RVI subcomponents, including baffle and former bolts, hold down springs, guide tube support pins, and other items surrounding the core. Every applicant, with support from the EPRI's Materials Reliability Project (MRP), has been involved in the process of defining this augmented inspection program. The issue technical group (ITG) under the auspices of the MRP is addressing this issue of RVI subcomponents' exposure to high neutron radiation and high temperature. Another issue, which is unique to the RVI, is the dimensional changes of those components exposed to very high neutron flux that could alter the coolant flow within the vessel. Due to these outstanding issues, every applicant has committed to address these aging effects when the industry understands and develops the necessary details of this new RVI inspection program.

Pressurizer

Scope

A pressurizer is connected to the RCS hot leg piping via the surge line and the cold leg piping via the spray line. The pressurizer is a vertical, cylindrical vessel with hemispherical top and bottom heads. The spray line and surge line nozzles have thermal sleeves to minimize thermal stresses in the line nozzles. Access is provided through a manway opening near the top of the pressurizer. During normal operation, the pressurizer contains a combination of borated reactor coolant and steam that is maintained at the desired temperature and pressure by electric heaters and the pressurizer spray system. The chemical and volume control system maintains the desired water level in the pressurizer during steady-state operation.

The pressurizer is designed to accommodate insurges and outsurges caused by power load transients and RCS temperature changes. During an insurge, the spray system condenses steam to prevent the pressure from reaching the operating point of the power-operated relief valves. A small continuous spray flow is provided to ensure that the water chemistry within the pressurizer is consistent with that in the RCS. During an outsurge, water flashes to steam due to the resulting pressure reduction, and the heaters are automatically actuated to keep the pressure above the minimum allowable limit.

Effects of Aging

The internal service environment for the pressurizer components is borated water. The environment for the external surfaces is the reactor building atmospheric condition. The aging effects applicable to the pressurizer components that require aging management are:

- cracking of carbon steel and low-alloy steel components in an air environment.
- cracking of stainless steel in a borated water environment.
- cracking and loss of material in nickel-based components in a borated water environment.
- loss of material from stainless steel components in a borated water environment.
- loss of material from carbon steel and low-alloy steel components in borated water leakage environment.
- loss of preload of the manway cover bolts/studs in the reactor building environment.

AMPs and TLAAs

The applicable AMPs for the pressurizer components are:

- Water Chemistry Control Program.
- Inservice Inspection Program.
- Boric Acid Corrosion Surveillance Program.
- Alloy 600 Aging Management Program.

The TLAA applicable to pressurizer components is metal fatigue.

Highlights of Review Findings

Pressurizers in many Westinghouse PWRs experience rapid thermal transients causing fatigue in the lower head and surge nozzle. The applicant must identify and describe the TLAA on metal fatigue for the pressurizer components. For TLAAs based on 10 CFR 54.21(c)(1), option (i) or (ii), the applicant must define the CUF and demonstrate that the TLAA meets the CLB fatigue design criterion, CUF < 1.0, for the period of extended operation. The fatigue evaluation should include the insurge/outsurge and other transient loads not included in the CLB, which are appropriate to such an extended TLAA, as described in the WOG report "Mitigation and Evaluation of Thermal Transients Caused by Insurges and Outsurges," MUHP-5060/5061/5062. The applicant must also consider the effects of the coolant environment on critical fatigue locations.

Intergranular Stress Corrosion Cracking (IGSCC) in the heat-affected zones of 304 stainless steel supports that are welded to the pressurizer cladding could grow as a result of thermal fatigue into the adjacent pressure boundary during the license renewal term. These welds will not require aging management in the period of extended operation if the applicant can provide reasonable justification that sensitization has not occurred in these welds during the fabrication of these components. Therefore, the applicant must provide a discussion of how the implementation of their plant-specific procedures and quality assurance requirements, if any, for the welding and testing of these austenitic stainless steel components provides reasonable assurance that sensitization has not occurred in these welds and associated heat-affected zones.

The staff notes that the potential to develop SCC in the materials for the manway bolts will be minimized if the yield strength of the material is held to less than 150 ksi, or the hardness is less than 32 on the Rockwell C hardness scale; however, the staff concludes that conformance with the minimum yield strength criteria in ASME Specification SA-193, Grade B7, does not in itself preclude a quenched and tempered low-alloy steel from developing SCC, especially if the acceptable yield strength is greater than the acceptable yield strength of 150 ksi. In accordance with EPRI NP-5769[22], the specification for this bolting material requires a minimum yield strength of 105 ksi. To take credit for the criteria in EPRI Report NP-5769, the applicant needs to verify the acceptable yield strengths for the quenched and tempered low-alloy steel bolting materials (e.g., SA-193 Grade B, materials) are in the range of 105-150 ksi.

In summary, the pressurizer components are subject to very high thermal cycles due to its design function of controlling the reactor pressure fluctuations. The spray and surge nozzles and associated attachments (e.g., thermal sleeves) have experienced cracking and other aging-related degradation during their current operations. Many components are fabricated from Alloy materials susceptible to PWSCC. The pressure relief valves and relief tanks also experience extreme conditions during normal plant operations. Similar to RCS piping and associated components, this RCS component contains many small-bore piping for instrumentation and control, and therefore, requires similar aging management. The welds connecting various pressurizer components, if sensitized during the welding process, may require an augmented inspection program for assessing their conditions.

Steam Generator

Scope

All steam generators (SGs), with the exception of B&W once-through steam generators (OTSGs), are vertical shell and U-tube evaporators with integral moisture separating equipment. Each OTSG is a vertical, straight tube, once-through, counterflow, shell-to-tube heat exchanger with shell-side boiling. Reactor coolant flows through the inverted U-tubes, entering and leaving through nozzles equipped with stainless steel safe ends located in the hemispherical bottom head of the steam generator. Steam is generated on the shell side of the tubes and the water-steam mixture flows upward through the tube bundle and into the steam drum section. Centrifugal moisture separators, located above the tube bundle, remove most of the entrained water from the steam. Steam dryers are employed to increase the steam quality before the steam flows upward to the outlet nozzle at the top of the steam generator.

Effects of Aging

The inside service environment for the primary water pressure boundary components (e.g., SG tubes, tubesheet) is borated water, while the outside environment of SG tubes is treated water. The environment for all other SG internal components is treated water. The environment of the SG external surfaces is the reactor building atmospheric condition. The aging effects applicable to the SG components that require aging management are:

- cracking in carbon steel, low-alloy steel, stainless steel (including cladding materials), nickel-based alloys (including buttering material) in borated and treated water environments.
- loss of material of carbon steel, low-alloy steel, stainless steel, and nickel-based alloys in both borated and treated water environment.
- loss of preload in low-alloy steel bolting in the reactor building air environment.

AMPs and TLAAs

The applicable AMPs for the steam generator components are:

- Water Chemistry Control Program
- Inservice Inspection Program
- Alloy 600 Aging Management Program
- Boric Acid Corrosion Surveillance Program
- Steam Generator Surveillance Program
- Flow Accelerated Corrosion Program

The TLAA applicable to steam generator components is metal fatigue.

Highlights of Review Findings

In accordance with the applicant's response to GL 97-06[23], "Degradation of Steam Generator Internals," all applicants indicated that they would follow the owners group recommendations as part of a continuing secondary-side inspection program. At that time, each applicant adopted appropriate inspection and maintenance activities to address all known degradation effects in steam generator internals. No near-term changes in the steam generator inspection programs were necessary. The owners groups suggested additional inspections to be included in the existing inspection activities for a long-term solution to the age-related degradation identified in GL 97-06.

The Steam Generator Surveillance Program, covers eddy current and visual examinations of Alloy 600 and Alloy 690 steam generator tubes, plugs, and sleeves, as well as internal support structures. The intent of the program is to detect loss of material and cracking prior to loss of intended function. Applicants have not included any SG internal support structures within the scope of the license renewal. Therefore, it is not clear whether the tubesheet, supports, and anti-vibration bars are inspected by this AMP. In addition, foreign objects search and retrieval (FOSAR) activities are conducted during the tube examination. The SG internals are periodically cleaned with sludge lancing or chemicals. Visual or video inspections of the tubesheet, tube support plates, tube bundles, wrapper, steam drum, feedwater distribution systems, and moisture separators are performed. SG welds are inspected as part of the inservice inspection program.

In summary, steam generators in nuclear power plants had a history of tube and tube support failures. Many plants have replaced their original SGs with new and improved SGs. These newer SG models are corrosion resistant and have better manufacturing techniques. Also, control of the water chemistry and adherence to plant technical specification have minimized degradation associated with SG subcomponents. It should be noted, however, that the current aging management activities concentrate on the primary water pressure boundary components (e.g., tubes) and tube supports. For long-term assurance that these pressure boundary components remain functional, applicants are incorporating programs and/ or activities to manage aging in other SG internal components.

Summary and Conclusions

The topical reports on RCS mechanical components developed by the owners groups and conditionally approved by the staff are found to be very comprehensive and provide the details of their assessments. The LRAs provide only the results of the scoping process and the AMRs performed on RCS components that require aging management for the period of extended operation. Typically, section 2 of the LRA provides a short description of the RCS components, section 3 presents the AMR results for the RCS components, and section 4 presents the applicable TLAAs and their evaluation results. Additional information is obtained from several appendices in the LRA covering the updated final safety analysis reports (UFSARs) and their supplements, evaluation of the AMPs in accordance with the ten attributes described in the NRC's SRP for license renewal, and AMR methodology (optional).

The aging management of the RCS mechanical components for the period of extended operation is demonstrated by using both existing and new aging management programs and activities in nuclear power plants. The existing programs include augmented plant-specific activities that are unique to the plant, and are based on operating and industry experience as well as NRC generic communications.

The Water Chemistry Control Program, the Inservice Inspection Program, the Alloy 600 Aging Management Program, and the Boric Acid Corrosion Surveillance Program are typically used to manage cracking and loss of material in RCS components. The Water Chemistry Control Program manages the relevant conditions that lead to the onset and propagation of loss of material and cracking that could lead to a loss of structure or component intended functions. The Inservice Inspection Program manages the cracking of weld locations, and the inspection, repair and replacement of pressure retaining components and their integral attachments. The Alloy 600 Aging Management Program manages cracking due to primary stress corrosion cracking (PWSCC) in components made out of Alloy 600/690, 82/182 and 52/152. The Boric Acid Corrosion Surveillance Program was developed in response to NRC Generic Letter 88-05[24], "Boric Acid Corrosion of Carbon Steel Reactor Pressure Boundary Components in PWR Plants," and manages loss of material in carbon steel components exposed to primary borated water. These programs are also applicable to other SSCs within the scope of license renewal for the plant.

The RCS component-specific programs that are identified for managing the effects aging include the Reactor Vessel Integrity Program, the Reactor Vessel Internals Inspection Program, and the Steam Generator Surveillance Program. The Reactor Vessel Internals Inspection Program is a new program which will monitor the conditions of reactor vessel internal components when exposed to both high radiation and high temperature conditions. The Reactor Vessel Integrity Program manages the reduction in fracture toughness of reactor vessel beltline materials to assure that the pressure boundary function of the reactor vessel beltline is maintained for the period of extended operation. This program complements the TLAA on Reactor Vessel Neutron Embrittlement which is required in accordance with 10 CFR Part 50, Section 50.60 (Appendix G, and Appendix H)[25], Section 50.61[26]. The Steam Generator Surveillance Program manages cracking in the steam generator tubes and other pressure boundary components. Also, plant-specific augmented inspections and monitoring activities are suggested by the licensee, depending on the generic communications with the industry as well as the NRC during the current licensing period. One applicant included several follow-up action items that will be performed as part of additional augmented aging management activities.

Based on plant-specific and industry operating experiences, individual plants have augmented their existing aging management activities with additional inspections of sensitized stainless steel materials for intergranular attack (IGA), reactor vessel incore flux thimble tubes for cracking, steam generator feedwater nozzles for cracking or erosion, reactor vessel head cracking, and pressurizer spray nozzle cracking.

Three aging effects, including cracking, loss of material, and loss of preload, will be managed using the Inservice Inspection Program and the Boric Acid Corrosion Surveillance Program. It is not clear how loss of preload will be managed by these two AMPs for the period of extended operation. None of these activities involve tightening of bolts where the loss of preload is observed. Similarly, applicants have proposed the Water Chemistry Control Program for management of the loss of material in certain RCS component. It is true that this program is important for controlling corrosion and other aging mechanisms that may lead to loss of material. However, to provide reasonable assurance that this program is effective, additional activities such as inspection or surveillance programs may be necessary.

There are follow-up actions for aging management activities which are being studied by applicants or by the EPRI-sponsored MRP-ITGs. These activities involves the additional inspections of reactor vessel internal components for both radiation and thermal aging, and thermal fatigue caused by cyclic thermal stratification and environmental effects. Other plant-specific issues include loss of preload in core barrel hold-down springs, augmented inspection of the pressurizer surge line connection to the hot-leg loop piping, and cracking of core support lugs.

In conclusion, all LRAs have adequately addressed the LR-related issues for their long-lived passive components within the RCS. However, the quality and the content of the LRAs has varied to some extent. Some applications provided details of the AMR process and AMR results, which were found to be adequate for the staff to develop reasonable assurance that these RCS components will be adequately managed during the period of extended operation. A few applications provided only the AMR results and did not refer to conditionally-approved owners group topical reports. In such cases, it became difficult for the staff to ensure that all applicable AMPs will adequately manage the aging effects identified by the AMR. Requests for additional information (RAIs) were written to obtain the information needed to complete the review of these LRAs.

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