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AUG 11 2014

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, DC 20555-0001

Salem Generating Station, Unit 1 & 2
Renewed Facility Operating License Nos. DPR-70 & DPR-75
NRC Docket Nos. 50-272 & 50-311

Subject: **Submittal of PWR Vessel Internals Inspection Plans for Aging Management of Reactor Internals at Salem Generating Station, Units 1 and 2**

References: (1) NUREG- 2101, "Safety Evaluation Report Related to the License Renewal of Salem Nuclear Generating Station", Appendix A, "Salem Generating Station License Renewal Commitments"
(2) Salem Generating Station Updated Final Safety Analysis Report (UFSAR), Rev. 26, dated May 21, 2012, Appendix B, Section A.5, "Salem License Renewal Commitment List"

Reference (1), Commitment 7, Sub-commitment 3 and Reference (2), Commitment 7, Sub-commitment 3 require PSEG Nuclear LLC to "...not less than 24 months before entering the period of extended operation, submit an Inspection Plan for reactor internals to the NRC for review and approval".

These requirements are met for both Salem Units 1 and 2 with the submittal of this letter and the attached two documents:

1. WCAP-17397-NP, Revision 2, "PWR Vessel Internals Program Plan for Aging Management of Reactor Internals at Salem Generating Station, Unit 1"
2. WCAP-17438-NP, Revision 2, "PWR Vessel Internals Program Plan for Aging Management of Reactor Internals at Salem Generating Station, Unit 2"

The attached documents are the reactor vessel internals inspection plans for Salem Units 1 & 2 respectively.

AUG 11 2014

There are no new regulatory commitments contained in this letter. If there are any questions, please contact John O'Rourke at 856-339-2142.

Sincerely,

John F. Perry

John F. Perry
Vice-President
Salem Generating Station

Attachments(2)

1. WCAP-17397-NP, Revision 2, "PWR Vessel Internals Program Plan for Aging Management of Reactor Internals at Salem Generating Station, Unit 1"
2. WCAP-17438-NP, Revision 2, "PWR Vessel Internals Program Plan for Aging Management of Reactor Internals at Salem Generating Station, Unit 2"

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Attachment 1

WCAP-17397-NP, Revision 2, "PWR Vessel Internals Program Plan for Aging Management of Reactor Internals at Salem Generating Station, Unit 1"

WCAP-17397-NP
Revision 2

July 2014

PWR Vessel Internals Program Plan for Aging Management of Reactor Internals at Salem Nuclear Generating Station, Salem Unit 1

Westinghouse

WCAP-17397-NP
Revision 2

PWR Vessel Internals Program Plan for Aging Management of Reactor Internals at Salem Nuclear Generating Station, Salem Unit 1

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LIST OF ACRONYMS

A/LAI	Applicant/Licensee Action Item
AMP	Aging Management Program Plan
AMR	Aging Management Review
ARDM	age-related degradation mechanism
ASME	American Society of Mechanical Engineers
B&PV	Boiler and Pressure Vessel
B&W	Babcock & Wilcox
BMI	bottom-mounted instrumentation
BWR	boiling water reactor
CASS	cast austenitic stainless steel
CE	Combustion Engineering
CFR	Code of Federal Regulations
CLB	current licensing basis
CMTR	certified material test report
CPE	corporate program engineer
CRGT	control rod guide tube
ECT	eddy current testing
EFPY	effective full-power years
EPRI	Electric Power Research Institute
ET	electromagnetic testing (eddy current)
EVT	enhanced visual testing (a visual NDE method that includes EVT-1)
FMECA	failure modes, effects, and criticality analysis
GALL	Generic Aging Lessons Learned
I&E	Inspection and Evaluation
IASCC	irradiation-assisted stress corrosion cracking
IE	irradiation embrittlement
INPO	Institute of Nuclear Power Operations
ISI	inservice inspection
ISR	irradiation-enhanced stress relaxation
LRA	license renewal application
LRAAI	license renewal applicant action items
MRP	Materials Reliability Program
NDE	nondestructive examination
NEI	Nuclear Energy Institute
NOS	Nuclear Oversight Section
NRC	Nuclear Regulatory Commission
NSSS	nuclear steam supply system
OE	Operating Experience
OEM	Original Equipment Manufacturer
OER	Operating Experience Report
PBD	program basis document
PH	precipitation-hardening
PSEG	Public Service Enterprise Group

LIST OF ACRONYMS (cont.)

PWR	pressurized water reactor
PWROG	Pressurized Water Reactor Owners Group
PWSCC	primary water stress corrosion cracking
QA	quality assurance
RCS	reactor coolant system
RIS	Regulatory Issue Summary
RO	refueling outage
RVI	reactor vessel internals
SCC	stress corrosion cracking
SE	Safety Evaluation
SER	Safety Evaluation Report
SGS	Salem Nuclear Generating Station
SPE	site program engineer
SRP	Standard Review Plan
SS	stainless steel
SSC	systems, structures, and components
TE	thermal embrittlement
TLAA	time-limited aging analysis
UFSAR	Updated Final Safety Analysis Report
U.S.	United States
UT	ultrasonic testing (a volumetric NDE method)
VT	visual testing (a visual NDE method that includes VT-1 and VT-3)
WOG	Westinghouse Owners Group
XL	extra-long Westinghouse fuel

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ACKNOWLEDGEMENTS

The authors would like to thank the members of the Public Services Enterprise Group, Aging Management Program Team led by Patrick Fabian, Samuel Speer, and our associates at Westinghouse for their efforts in supporting development of this WCAP.

1 PURPOSE

The purpose of this report is to document the Salem Nuclear Generating Station (SGS), Unit 1, hereafter SGS Unit 1, Reactor Internals Aging Management Program Plan (AMP). This revision updates the report to incorporate the latest revision of Reference [46].

Public Services Enterprise Group (PSEG) Nuclear LLC is a Delaware limited liability company formed to own and operate nuclear generating stations. PSEG Nuclear LLC is a wholly owned subsidiary of PSEG Power LLC, which is wholly owned by PSEG Incorporated, a corporation formed under the laws of the State of New Jersey. PSEG Nuclear LLC is the licensed operator of SGS Unit 1.

The purpose of the AMP is to manage the effects of aging on the reactor internals of SGS Unit 1 through the period of extended operation. SGS Unit 1 enters the period of extended operation at midnight on August 13, 2016 [1]. This document provides a description of the program as it relates to the management of aging effects identified in various regulatory and updated industry-generated documents, in addition to the program documented in SGS Unit 1 procedure document ER-AP-333 [2], for reactor internals. It is prepared in accordance with the various regulatory and industry-generated documents, and is supported by existing SGS documents and procedures. As required, by industry experience or directive in the future, it will be updated or supported by additional documents to provide clear and concise direction for the effective management of aging degradation in reactor internals components. These actions provide assurance that operations at SGS Unit 1 will continue to be conducted in accordance with the current licensing basis (CLB) for the reactor internals by fulfilling license renewal commitments [3], United States (U.S.) Nuclear Regulatory Commission (NRC) expectations in the Regulatory Issue Summary (RIS) [4], following American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel (B&PV) Code Section XI Inservice Inspection (ISI) requirements [5], and meeting industry requirements [6]. This AMP fully captures the intent of the additional industry guidance for reactor internals augmented inspections, based on the programs sponsored by U.S. utilities through the Electric Power Research Institute (EPRI) managed Materials Reliability Program (MRP) and the Pressurized Water Reactor Owners Group (PWROG).

The main objectives for the SGS Unit 1 reactor vessel internals (RVI) AMP are to:

- Demonstrate that the effects of aging on the reactor internals will be adequately managed for the period of extended operation in accordance with 10 CFR 54 [7].
- Summarize the role of existing SGS Unit 1 aging management programs in the reactor vessel internals AMP.
- Define and implement the industry-defined (EPRI/MRP and PWROG) pressurized water reactor (PWR) reactor vessel internals requirements and guidance for managing aging effects on reactor internals.
- Provide an inspection plan summary for the SGS Unit 1 reactor internals.

The Safety Evaluation Report (SER) for SGS Unit 1 license renewal [3] includes the license renewal commitment for the reactor vessel internals. This commitment is also documented in the SGS aging

management program basis document (PBD) [8] and SGS Updated Final Safety Analysis Report (UFSAR) Appendix B, Section A.2.1.7, "PWR Vessel Internals" [9].

SGS License Renewal Commitment 7 addresses the creation of a program for the PWR Reactor Vessel Internals at SGS Unit 1.

Commitment 7 [3]: PWR Vessel Internals is a new program that will include the following activities:

1. Participate in the industry programs for investigating and managing aging effects on reactor internals.
2. Evaluate and implement the results of the industry programs as applicable to the reactor internals.
3. Upon completion of these programs, but not less than 24 months before entering the period of extended operation, submit an inspection plan for reactor internals to the NRC for review and approval.

The SER also included commitments related to reactor vessel internals. The purpose of these commitments is to continue the ongoing programs for ASME XI ISIs, Subsections IWB, IWC, and IWD (License Renewal Commitment 1) and water chemistry (License Renewal Commitment 2) and to implement a new program for flux thimble tube inspection (License Renewal Commitment 25).

Augmented inspections, based on required program enhancements resulting from industry programs [6], will become part of the SGS Unit 1 ISI program [10]. Corrective actions for augmented inspections will be developed and will use repair and replacement procedures equivalent to those requirements in ASME B&PV Code, Section XI [5], or they will use processes determined to be equivalent to or more rigorous than currently defined procedures as determined independently by PSEG Nuclear, or in cooperation with the industry.

This AMP for the SGS Unit 1 reactor internals demonstrates that the program adequately manages the effects of aging for reactor internals components. The AMP establishes the basis for providing reasonable assurance that the internals components will continue to perform their intended function through the SGS Unit 1 period of extended operation. This AMP supports the SGS Unit 1 License Renewal Commitment 7 to submit a PWR Vessel Internals inspection plan to the NRC, as it will be implemented from PSEG Nuclear participation in industry initiatives, 24 months prior to the period of extended operation. Thus, the program must be submitted no later than August 13, 2014.

The development and implementation of this program meets the guidelines provided in the RIS [4] by supporting the commitment to submit a PWR Vessel Internals inspection plan in accordance with MRP-227-A for SGS Unit 1.

2 BACKGROUND

The management of aging degradation effects in reactor internals is required for nuclear plants considering or entering license renewal, as specified in the NRC Standard Review Plan (SRP) [11]. The U.S. nuclear power industry has been actively engaged in recent years in a significant effort to support the industry goal of responding to these requirements. Various programs have been underway within the industry over the past decade to develop guidelines for managing the effects of aging within PWR reactor internals. In 1997, the Westinghouse Owners Group (formerly WOG, now PWROG) issued WCAP-14577 [12], "License Renewal Evaluation: Aging Management for Reactor Internals," which was reissued as Revision 1-A in 2001 after receiving NRC staff review and approval. Later, an effort was engaged by the EPRI MRP to address the PWR internals aging management issue for the three currently operating U.S. reactor designs - Westinghouse, Combustion Engineering (CE), and Babcock & Wilcox (B&W).

The MRP first established a framework and strategy for the aging management of PWR internals components using proven and familiar methods for inspection, monitoring, surveillance, and communication. Based upon that framework and strategy, as well as accumulated industry research data, the following elements of an AMP were further developed [12, 14]:

- Screening criteria were developed, considering chemical composition, neutron fluence exposure, temperature history, and representative stress levels, for determining the relative susceptibility of PWR internals components to each of eight postulated aging mechanisms (further discussed in Section 4 of this program).

- PWR internals components were categorized, based on the screening criteria, into categories that ranged from:

  - Components for which the effects from the postulated aging mechanisms are insignificant,
  - Components that are moderately susceptible to the aging effects, and
  - Components that are significantly susceptible to the aging effects.

- Functionality assessments were performed to determine the effects of the degradation mechanisms on component functionality. These assessments were based on representative plant designs of PWR internals components and assemblies of components using irradiated and aged material properties.

Aging management strategies were developed by combining the results of the functionality assessment with several contributing factors to determine the appropriate aging management methodology, baseline examination timing, and the need and timing of subsequent inspections. Items considered included component accessibility, operating experience (OE), existing evaluations, and prior examination results.

The industry guidance is contained within two separate EPRI MRP documents:

- MRP-227-A [6], "Pressurized Water Reactor Internals Inspection and Evaluation Guidelines," (hereafter referred to as "the Inspection and Evaluation (I&E) Guidelines" or simply "MRP-227-A") provides industry background, listing of reactor internals components requiring inspection,

type of nondestructive examination (NDE) required for each component, timing for initial inspections, and criteria for evaluating inspection results. MRP-227-A provides a standardized approach to PWR internals aging management for each unique reactor design (Westinghouse, B&W, and CE).

- MRP-228, Revision 1 [15], "Inspection Standard for PWR Internals," provides guidance on the qualification/demonstration of the required NDE techniques and other criteria pertaining to the actual performance of the inspections.

The PWROG has developed WCAP-17096-NP, Revision 2, "Reactor Internals Acceptance Criteria Methodology and Data Requirements" for the MRP-227, Revision 0 components, where feasible [16]. This document has been submitted to the NRC for review and approval, and will be updated to incorporate changes from MRP-227-A [6]. Final reports are to be developed and available for industry use in support of planned license renewal inspection commitments. In some cases, individual plants will develop plant-specific acceptance criteria for some internals components where a generic approach is not practical.

The SGS reactor internals for SGS Unit 1 are integral with the reactor coolant system (RCS) of a Westinghouse four-loop nuclear steam supply system (NSSS). Illustrations of typical reactor internals are provided in Figures A-1 through A-14.

As described in the SGS license renewal application (LRA) [1], the reactor vessel internals consist of the upper core support structure, the lower core support structure, and the in-core instrumentation support structure, where each of these major components has a distinct purpose. Also included are the flux thimble tubes that extend from the penetrations on the reactor vessel lower head up to the seal table. In addition, the major structural welds that form or join the major structures, the minor structural welds joining parts such as lifting lugs, supports, and tubes to the major structures, and the fasteners and alignment pins that guide, align, and fasten the major structures are within the scope of the reactor vessel internals. The reactor vessel internals also include the fuel assemblies and the rod cluster control assemblies that are supported by all three structures; however, these are subject to replacement in accordance with the Reload Control Process and as such, they are short-lived components and are not subject to the aging management requirements of MRP-227-A.

The upper core support structure consists of the upper support assembly, the upper core plate, support columns, and the control rod guide assemblies. The support columns establish the spacing between the upper support assembly and the upper core plate. The upper core plate consists of openings for the control rod guide tubes, and for the distribution of reactor coolant flow via orifice plates, integral flow mixers, and open holes. The control rod guide tube assemblies shield and guide the control rod drive shafts and control rods. They are fastened to the upper core plate and are guided by pins into the upper core plate for proper orientation and support. A large circumferential hold down spring restrains axial movements. The entire upper core support structure is removed as a unit during refueling operations to permit access to the fuel assemblies.

The purpose of the upper core support structure is to contain the guide tube assemblies that shield and guide the control rod drive shafts and control rods. This structure engages the top of the fuel assemblies and provides structural support experienced by transverse loadings from coolant crossflow and other

design conditions. The upper core support structure also provides structural support for vertical loads from the fuel, hydraulic forces, control rod dynamics, and other design loadings.

The lower core support structure remains in place in the reactor vessel during most refueling operations and is only removed to perform scheduled inspections. The lower core support structure is supported at its upper flange from a ledge in the reactor vessel and its lower end is restrained from transverse motion by a radial support system attached to the reactor vessel wall. The lower core support structure consists of the core barrel, the core baffles, the flow distribution (diffuser) plate, the lower core plate and support columns, the thermal shield, and the core support forging, which is welded to the core barrel. The core barrel supports and contains the fuel assemblies. The core barrel directs coolant flow upwards through the reactor vessel by means of the bottom-mounted flow distribution plate and the core baffles. The lower core support plate provides support for the support columns, reactor coolant flow distribution, and support and orientation of the fuel assemblies. The one-piece thermal shield provides neutron shielding when fuel is present in the core, and is fixed to the core barrel at the top with rigid bolted connections. The bottom of the thermal shield is connected to the core barrel by means of axial flexures. This bottom support allows for differential axial growth of the shield/core barrel but restricts radial or horizontal movement of the bottom of the thermal shield. Specimen guides are welded to the thermal shield and allow for irradiation of test samples during operations. The core support is contoured to the bottom of the reactor vessel and receives the weight, hydraulic, and control rod dynamic loadings.

The purpose of the lower core support structure is to form a periphery enclosure of the core including core baffles and a bottom flow distribution plate for efficient flow distribution, provide neutron shielding by means of the thermal shield, and to provide structural support experienced by transverse loadings from coolant crossflow and other design conditions. The lower core support structure also provides structural support for vertical loads from the fuel, hydraulic forces, control rod dynamics, and other design loadings.

The purpose of the in-core instrumentation support structure is to provide structural support for the bottom-mounted in-core instrumentation (flux thimbles and thermocouples) and to maintain a pressure boundary between the reactor coolant and containment atmosphere.

The SGS LRA lists the following system intended functions for the reactor vessel internals [1]:

1. Maintain reactor core assembly geometry. The reactor vessel internals maintains core assembly geometry within the reactor to ensure core cooling, core reactivity control, and the integrity of the fuel cladding as a radioactive material barrier. 10 CFR 54.4(a)(1)

2. Achieve and maintain the reactor core subcritical for any mode of normal operation or event. The rod cluster control assemblies adjust the concentration of the neutron absorber in the core. 10 CFR 54.4(a)(1)

3. Introduce emergency negative reactivity to make the reactor subcritical. Following a reactor trip signal, all rod cluster control assemblies are released into the core to initiate a complete reactor trip. 10 CFR 54.4(a)(1)

4. Resist nonsafety-related systems, structures, and components (SSCs) failure that could prevent satisfactory accomplishment of a safety-related function. The control rods are non-safety-related components that have the potential for spatial interactions with safety-related SSCs. 10 CFR 54.4(a)(2)

SGS Unit 1 was granted a license for extended operation by the NRC through the issuance of a SER in NUREG-2101 [3]. In the SER, the NRC concluded that the reactor vessel internal systems, structures, and components that are subject to an Aging Management Review (AMR) had been adequately identified, as required by 10 CFR 54.21(a)(3) and that the requirements of 10 CFR 54.29 [7] have been met. A listing of the SGS Unit 1 reactor vessel internals components and subcomponents that are subject to AMP requirements is included in Table B-1.

In accordance with 10 CFR Part 54 [7], frequently referred to as the License Renewal Rule, SGS has developed a PBD to manage the aging of reactor vessel internals components and structures in accordance with NUREG-1801, XI.M16 [8]. The U.S. nuclear industry, through the efforts of the MRP and PWROG, has further investigated the components and subcomponents that require aging management to support continued reliable function. As designated by the Nuclear Energy Institute (NEI) protocols in NEI 03-08 [17], "Guidelines for the Management of Materials Issues," each plant will be required to use MRP-227-A and MRP-228 to develop and implement an AMP for reactor internals no later than three years after the initial industry issuance of MRP-227, Revision 0. MRP-227, Revision 0 was issued in December 2008, and plant AMPs must therefore be completed by December 2011 or sooner, as required by plant-specific license renewal commitments. Revision 0 of this AMP was completed to satisfy this MRP-227 requirement. According to [4], SGS Unit 1 is a Category B plant that is expected to submit their RVI AMP/inspection plan based on the guidance of MRP-227-A, consistent with their commitments. Per the LRA [1], SGS Unit 1 has a commitment to submit an inspection plan for reactor internals for approval by the NRC no later than August 13, 2014.

The information contained in this AMP fully complies with the requirements and guidance of the referenced documents. The AMP will manage aging effects of the RVI so that the intended functions will be maintained consistent with the CLB for the period of extended operation.

3 PROGRAM OWNER

The PWR Vessel Internals Program [2] manages the effects of age-related degradation mechanisms of reactor vessel internals. The successful implementation and comprehensive long-term management of the SGS reactor vessel internals AMP will require the integration of PSEG Nuclear organizations, corporately and at SGS, and interaction with multiple industry organizations including, but not limited to, the ASME, MRP, NRC, and PWROG. The responsibilities of the individual PSEG Nuclear corporate and SGS groups are provided in the following paragraphs. PSEG Nuclear will maintain cognizance of industry activities related to PWR internals inspection and aging management, and will address/implement industry guidance stemming from those activities, as appropriate under NEI 03-08 practices.

The overall responsibility for administration of the reactor vessel internals AMP is SGS Unit 1 senior management.

Additional responsibilities and the appropriate responsible personnel, as described in Section 3 of [2], are discussed in the following subsections.

3.1 MANAGER OF ENGINEERING PROGRAMS

The manager of engineering programs is responsible for the overall implementation of the PWR Reactor Internals Inspection program.

3.2 SITE PROGRAM ENGINEER (SPE)

The SPE is responsible for executing the PWR Reactor Internals Inspection program per the requirements of MRP-227-A [6].

3.3 CORPORATE PROGRAM ENGINEER (CPE)

The CPE is responsible for governance and oversight of PWR Reactor Internals Inspection program.

3.4 NDE SERVICES INSPECTOR/EXAMINER

The NDE services inspector/examiner is responsible for inspection of PWR Reactor Internals per the requirements of MRP-227-A and MRP-228 [15].

4 DESCRIPTION OF SGS REACTOR INTERNALS AGING MANAGEMENT PROGRAMS AND INDUSTRY PROGRAMS

The U.S. nuclear industry, through the combined efforts of utilities, vendors, and independent consultants, has defined a generic guideline to assist utilities in developing reactor internals plant-specific AMPs based on inspection and evaluation. The intent of the SGS Unit 1 AMP is to ensure the long-term integrity and safe operation of the reactor internals components. SGS has developed this AMP in conformance with the 10 Generic Aging Lessons Learned (GALL) [18] attributes and MRP-227-A [6].

This reactor internals AMP utilizes a combination of prevention, mitigation, and condition monitoring. Where applicable, credit is taken for existing programs such as water chemistry [19, 20] and inspections prescribed by the ASME Section XI ISI Program [10], and past and future mitigation projects such as control rod guide tube support pin replacement [38] and flux thimble tube replacement and inspection [39]. These existing programs are augmented with the inspections and evaluations recommended by MRP-227-A.

Aging degradation mechanisms that impact internals have been identified and documented in the LRA submitted by PSEG Nuclear [1]. The overall outcome of the reviews and the additional work performed by the industry, as summarized in MRP-227-A, is to provide appropriate augmented inspections for reactor internals components to provide early detection of the degradation mechanisms of concern. Therefore, this AMP is consistent with the existing SGS Unit 1 reactor vessel internals AMR methodology [8] and the additional industry work summarized in MRP-227-A. All sources are consistent and address concerns about component degradation resulting from the following eight material aging degradation mechanisms identified as affecting reactor internals:

- Stress Corrosion Cracking

  Stress corrosion cracking (SCC) refers to local, non-ductile cracking of a material due to a combination of tensile stress, environment, and metallurgical properties. In primary water, this may be referred to as PWSCC. The actual mechanism that causes SCC or PWSCC involves a complex interaction of environmental and metallurgical factors. The aging effect is cracking.

- Irradiation-Assisted Stress Corrosion Cracking

  Irradiation-assisted stress corrosion cracking (IASCC) is a unique form of SCC that occurs only in highly-irradiated components. The aging effect is cracking.

- Wear

  Wear is caused by the relative motion between adjacent surfaces, with the extent determined by the relative properties of the adjacent materials and their surface condition. The aging effect is loss of material.

- Fatigue

Fatigue is defined as the structural deterioration that can occur as the result of repeated stress/strain cycles caused by fluctuating loads and temperatures. After repeated cyclic loading of sufficient magnitude, microstructural damage can accumulate, leading to macroscopic crack initiation at the most highly affected locations. Subsequent mechanical or thermal cyclic loading can lead to growth of the initiated crack. Corrosion fatigue is included in the degradation description.

Low-cycle fatigue is defined as cyclic loads that cause significant plastic strain in the highly stressed regions, where the number of applied cycles is increased to the point where the crack eventually initiates. When the cyclic loads are such that significant plastic deformation does not occur in the highly stressed regions, but the loads are of such increased frequency that a fatigue crack eventually initiates, the damage accumulated is said to have been caused by high-cycle fatigue. The aging effects of low-cycle fatigue and high-cycle fatigue are additive.

Fatigue crack initiation and growth resistance are governed by a number of material, structural, and environmental factors such as stress range, loading frequency, surface condition, and presence of deleterious chemical species. Cracks typically initiate at local geometric stress concentrations such as notches, surface defects, and structural discontinuities. The aging effect is cracking.

- Thermal Aging Embrittlement

Thermal aging embrittlement is the exposure of delta ferrite within cast austenitic stainless steel (CASS) and precipitation-hardening (PH) stainless steel to high inservice temperatures, which can result in an increase in tensile strength, a decrease in ductility, and a loss of fracture toughness. Some degree of thermal aging embrittlement can also occur at normal operating temperatures for CASS and PH stainless steel internals. CASS components have a duplex microstructure and are particularly susceptible to this mechanism. While the initial aging effect is loss of ductility and toughness, unstable crack extension is the eventual aging effect if a crack is present and the local applied stress intensity exceeds the reduced fracture toughness.

- Irradiation Embrittlement

Irradiation embrittlement is also referred to as neutron embrittlement. When exposed to high-energy neutrons, the mechanical properties of stainless steel and nickel-based alloys can be changed. Such changes in mechanical properties include increasing yield strength, increasing ultimate strength, decreasing ductility, and a loss of fracture toughness. The irradiation embrittlement aging mechanism is a function of both temperature and neutron fluence. While the initial aging effect is loss of ductility and toughness, unstable crack extension is the eventual aging effect if a crack is present and the local applied stress intensity exceeds the reduced fracture toughness.

- Void Swelling and Irradiation Growth

  Void swelling is defined as a gradual increase in the volume of a component caused by formation of microscopic cavities in the material. These cavities result from the nucleation and growth of clusters of irradiation-produced vacancies. Helium produced by nuclear transmutations can have a significant impact on the nucleation and growth of cavities in the material. Void swelling may produce dimensional changes that exceed the tolerances on a component. Strain gradients produced by differential swelling in the system may produce significant stresses. Severe swelling (>5 percent by volume) has been correlated with extremely low fracture toughness values. Also included in this mechanism is irradiation growth of anisotropic materials, which is known to cause significant dimensional changes within in-core instrumentation tubes that are fabricated from zirconium alloys. While the initial aging effect is dimensional change and distortion, severe void swelling may result in cracking under stress.

- Thermal and Irradiation-Enhanced Stress Relaxation or Irradiation-Enhanced Creep

  The loss of preload aging effect can be caused by the aging mechanisms of stress relaxation or creep. Thermal stress relaxation (or primary creep) is defined as the unloading of preloaded components due to long-term exposure to elevated temperatures, as seen in PWR internals. Stress relaxation occurs under conditions of constant strain where part of the elastic strain is replaced with plastic strain. Available data show that thermal stress relaxation appears to reach saturation in a short time (< 1000 hours) at PWR internals temperatures.

  Creep (or more precisely, secondary creep) is a slow, time- and temperature-dependent, plastic deformation of materials that can occur when subjected to stress levels below the yield strength (elastic limit). Creep occurs at elevated temperatures where continuous deformation takes place under constant strain. Secondary creep in austenitic stainless steels is associated with temperatures higher than those relevant to PWR internals even after taking into account gamma heating. However, irradiation-enhanced creep (or more simply, irradiation creep) or irradiation-enhanced stress relaxation (ISR) is an athermal process that depends on the neutron fluence and stress, and it can also be affected by void swelling should it occur. The aging effect is a loss of mechanical closure integrity (or, preload) that can lead to unanticipated loading that, in turn, may eventually cause subsequent degradation by fatigue or wear and result in cracking.

The SGS Unit 1 RVI AMP is focused on meeting the requirements of the 10 elements of an aging management program as described in NUREG-1801, GALL Report Section XI.M16A for PWR Vessel Internals. In the SGS Unit 1 RVI AMP, this is demonstrated through application of existing SGS AMR methodology that credits inspections prescribed by the ASME Code Section XI ISI Program, existing SGS programs, and additional augmented inspections based on MRP-227-A recommendations. A description of the applicable existing SGS programs and compliance with the elements of the GALL is contained in the following subsections.

4.1 SGS PROGRAMS

SGS's overall strategy for managing aging in reactor internals components is supported by the following existing programs [1]:

- Primary Water Chemistry Program [19, 20]
- ASME Section XI ISI Program [10]
- Metal Fatigue of Reactor Coolant Pressure Boundary [40]

These are established programs that support the aging management of RCS components in addition to the reactor vessel internals components. Although affiliated with and supporting the reactor vessel internals AMP, these programs will continue to be managed under the existing structure.

SGS's overall strategy for managing aging in reactor internals components will also be supported by the new reactor vessel internals aging management activities (see Section 4.2.2), flux thimble tube inspections, and the metal fatigue of reactor pressure boundary program.

Brief descriptions of the programs are included in the following subsections.

4.1.1 Primary Water Chemistry Program

The SGS Unit 1 Primary Strategic Water Chemistry Plan is an existing program [19, 20] that provides activities for monitoring and controlling the chemical environments of the SGS primary cycle systems such that aging effects of system components are minimized. This program manages the aging effects of cracking, loss of material, reduction of neutron-absorbing capacity, and reduction of heat transfer. The program mitigates damage caused by corrosion and SCC and other aging mechanisms. This program includes provisions specified by NUREG-1801 for the verification of proper chemistry control and aging management, such that the intended functions of plant components will be maintained during the period of extended operation for SGS Unit 1.

The SGS Unit 1 water chemistry aging management program includes periodic sampling of primary water for the known detrimental contaminants specified in the EPRI PWR water chemistry guidelines to maintain their concentrations below levels known to result in loss of material or cracking. Sampling frequencies and action limits for each control parameter are defined in SGS-specific procedures.

SGS follows the guidance set forth in the EPRI PWR Primary Water Chemistry Guidelines [13], which is referenced in NUREG-1801, XI.M2 (which refers to Revision 3 of the guidelines - EPRI TR-105714). Later revisions of the guidelines will be used when issued. The limits imposed by the SGS Program meet the intent of the industry standard for addressing primary water chemistry [13] and includes SGS Unit 1 Plant UFSAR [9] limits for specific chemical control parameters.

The evaluation of this program against the 10 attributes in the GALL for Program XI.M2 in support of the SGS LRA remains applicable.

4.1.2 ASME Section XI Inservice Inspection Program

The ASME Code Section XI Inservice Program is part of the SGS ISI Program [10]. This existing program includes inspections that are performed to manage aging effects such as cracking, loss of fracture toughness and loss of material in Class 1, 2, and 3, piping and components exposed to air, reactor coolant, steam, treated water and treated borated water environments within the scope of license renewal. The SGS ASME Section XI ISI Program is augmented, as identified in the SGS LRA [1], to also manage effects of aging by other programs. For SGS Unit 1, inspections conducted under the reactor vessel

internals AMP will be controlled as a combination of ASME Code Section XI ISI examinations on core support structures and augmented examinations performed under the ISI Program for the reactor vessel internals components addressed within MRP-227-A. ASME Code Section XI, 10-year ISI examinations supporting the period of extended operation are currently scheduled for the 1RF26 (Spring 2019).

The evaluation of this program against the 10 attributes in the GALL for Program XI.M1 in support of the SGS LRA remains applicable.

4.1.3 Metal Fatigue of Reactor Pressure Boundary Program

The SGS Metal Fatigue of Reactor Pressure Boundary Program [40] is an existing program that manages cumulative fatigue usage for the reactor vessel, the pressurizer, the steam generators, Class 1 and non-Class 1 piping, and Class 1 components subject to the reactor coolant, treated borated water, and treated water environments. The Metal Fatigue of Reactor Pressure Boundary Program is a preventive program that monitors and tracks the number of critical thermal and pressure transients to ensure that the cumulative usage factors for selected reactor coolant system (RCS) components remain less than 1.00 through the period of extended operation. The program determines the number of transients that occur and updates the 60-year projections as required on an annual basis.

The evaluation of this program against the 10 attributes in the GALL for Program X.M1 in support of the SGS LRA remains applicable.

4.2 SUPPORTING SALEM NUCLEAR GENERATING STATION UNIT 1 PROGRAMS AND AGING MANAGEMENT SUPPORTIVE PLANT ENHANCEMENTS

4.2.1 Reactor Internals Aging Management Review Process

A comprehensive review of aging management of SGS Unit 1 reactor internals was performed according to the requirements of the License Renewal Rule [7]. This review was conducted in support of the SGS LRA [1]. The SGS Unit 1 LRA was approved by the NRC in NUREG-2101 [3]. The SGS LRA, subsection 2.3.1.3, Table 3.1.1 and Table 3.1.2-3 identified the reactor vessel internals components that are subject to AMR. The LRA Table 3.1.2-3 Reactor Vessel Internals – Summary of Aging Management Evaluation provides the detailed results of the AMR conducted on the reactor vessel internals components. It also includes a comparison to the relevant NUREG-1801, Volume 2 item to note consistencies and there are footnotes to explain exceptions.

The AMR supported the SGS LRA as follows:

1. Identified applicable aging effects requiring management.

2. Evaluated existing aging management programs and commitments to ensure that they adequately manage those aging effects.

3. Identified actions for modifications to existing programs or actions to create new aging management programs, and any other actions required to support the conclusions reached in the review.

AMRs were performed for each SGS Unit 1 system that contained long-lived, passive components requiring AMR and the results are incorporated into the SGS LRA. This review is not repeated here, but the results are fully incorporated into the SGS Unit 1 RVI AMP. It should be noted that some RVI components listed in the SGS Unit 1 AMR were not plant specific. The non-applicable components have been noted within Table B-1.

4.2.2 Pressurized Water Reactor Internals Aging Management Program

The PWR Internals aging management program [2] is a new program that provides reasonable assurance that the changes in dimensions, cracking, loss of fracture toughness, and loss of preload aging effects is adequately managed so that the intended functions of components within the scope of license renewal is maintained consistent with the current licensing basis during the period of extended operation.

The procedure [2] establishes the PWR Vessel Internals aging management program for the inspection, repair, replacement, degradation evaluation, and mitigation of the PWR Internals, which ensures that MRP guidelines are met. The PWR Vessel Internals program establishes a framework and structure to existing PWR Vessel Internals efforts and implements the guidelines of MRP-227-A [6] and MRP-228 [15].

The reactor coolant system CASS components are maintained by inspecting and evaluating the extent of thermal aging embrittlement in accordance with the requirements of the ASME Boiler and Pressure Vessel Code, Section XI. The SGS Unit 1 PWR Vessel Internals Program will be used to manage the loss of fracture toughness due to thermal aging and neutron irradiation embrittlement of CASS reactor vessel internals components exposed to reactor coolant and neutron flux.

4.2.3 Flux Thimble Tube Inspection

Flux thimble tubes are long, slender, stainless steel tubes that are seal welded at one end with flux thimble tube plugs, which pass through the vessel penetration, through the lower internals assembly, and finally extend to the top of the fuel assembly. The bottom-mounted instrumentation (BMI) column assemblies provide a path for the flux thimbles into the core from the bottom of the vessel and protect the flux thimbles during operation of the reactor. The flux thimble provides a path for the neutron flux detector into the core and is subject to reactor coolant pressure on the outside and containment pressure on the inside.

The SGS Flux Thimble Tube Inspection Program [39] is a new program that manages the loss of material due to wear of the flux thimble tube materials. It implements the recommendations of NRC Bulletin 88-09 [21] that a tube wear inspection procedure be established and maintained for Westinghouse-supplied reactors that use bottom-mounted flux thimble tube instrumentation. The program utilizes an inspection methodology such as ECT to inspect the flux thimble tubes on a periodic frequency to monitor wall thinning and predict when tubes would require repair or replacement. The program implements a wall thickness trending report.

The Flux Thimble Tube Inspection program establishes appropriate acceptance criteria (percent through-wall wear), based on industry guidance and including margin to include allowances for factors such as instrument uncertainty, uncertainties in wear scar geometry, and other potential inaccuracies, as applicable, to the inspection methodology.

Previously, a Flux Thimble Tube Inspection was in effect from 1985 to 1993. SGS Unit 1 replaced all of the flux thimble tubes during the 1RF7 (1990) outage with an improved design. 1RF10 (1993) activities for SGS Unit 1 included ECT of eleven of the improved design flux thimble tubes. The results indicated that there was no significant wear on any of the eleven inspected flux thimble tubes. Indications found were attributed to incomplete tube cut scar and a partial tube cut. Also, the examinations indicated that there was no cladding bulging or ovality detected. As a result of the examinations during 1RF10 (1993), SGS notified the NRC that it would discontinue future periodic inspections of the flux thimble tubes.

The re-implementation of the Flux Thimble Tube Inspection Program is consistent with the 10 elements of the aging management program XI.M37, Flux Thimble Tube Inspection, specified in NUREG-1801. The examples from the past program demonstrate the effectiveness of the program, where improved design changes were made to replace the flux thimble tube materials with those of an improved design and were later inspected to prove that there was minimal wear.

The evaluation of this program against the 10 attributes in the GALL for Program XI.M37 in support of the SGS LRA remains applicable.

4.2.4 Control Rod Guide Tube Support Pin Replacement Project

The control rod guide tube support pins are used to align the bottom of the control rod guide tube assembly into the top of the upper core plate. In general, SCC prevention is aided by adherence to strict primary water chemistry limits that effectively mitigate SCC and greatly reduce the probability of IASCC. The limits imposed by the Primary Water Chemistry Program at SGS Unit 1 are consistent with the latest EPRI guidelines as described in Section 4.1.

The original SGS Unit 1 support pins were fabricated from *INCONEL*® Alloy X-750 that was hot rolled, solution treated, and age hardened at various temperatures and times depending on heat, manufacturer, and fabrication date. Support pins made of this material with the associated heat treatments were shown to be susceptible to PWSCC and likely to fail during the lifetime of a nuclear power plant. Westinghouse developed an improved support pin design and fabrication technique that significantly reduced the susceptibility to PWSCC while maintaining the fatigue and wear requirements necessary to support continued uninterrupted service [22].

In response to the industry concern, the support pins were replaced at SGS Unit 1 in the spring of 2007; the replacement support pins utilized improved materials (strain-hardened 316 stainless steel) that support the proactive management of aging in reactor internals components. Detailed descriptions of the replacement are retained in the plant records [38].

4.2.5 Reactor Vessel Internals Fatigue Analyses

The SGS Unit 1 reactor vessel internals were implicitly designed for low-cycle fatigue based upon the reactor coolant system design transient projection for 40 years; this is identified as a time-limiting aging analysis (TLAA). Post-design analyses consist of two Westinghouse analyses; (1) a lower core plate evaluation based on the 1.4% uprate and (2) qualification of the SGS Unit 1 domed lower core support plate, also part of the 1.4% uprate project. The effect of the 1.4% uprate was deemed negligible; therefore, cumulative fatigue usage attributable to the thermal design transients did not change [1]. SGS Unit 1 monitors and counts the plant operational cycles [40]. If the 60-year projected number of cycles is less than the number of cycles used in the design fatigue analysis, then the fatigue analyses based upon the design transients will remain valid for 60 years of operation if the design transient severity also bounds the actual transient severity.

The calculations and transient monitoring program demonstrate that the effects of aging on the reactor vessel internals will be adequately managed for the period of extended operation in accordance with 10 CFR 54.21(c)(1)(i).

4.3 INDUSTRY PROGRAMS

4.3.1 MRP-227-A, Reactor Internals Inspection and Evaluation Guidelines

MRP-227-A [6], as discussed in Section 2, was developed by a team of industry experts including utility representatives, NSSS vendors, independent consultants, and international committee representatives who reviewed available data and industry experience on materials aging. The objective of the group was to develop a consistent, systematic approach for identifying and prioritizing inspection and evaluation requirements for reactor internals. The following subsections briefly describe the industry process.

4.3.1.1 MRP-227-A Reactor Vessel Internals Component Categorizations

MRP-227-A used a screening and ranking process to aid in the identification of required inspections for specific RVI components. MRP-227-A credited existing component inspections, when they were deemed adequate, as a result of detailed expert panel assessments conducted in conjunction with the development of the industry document. Through the elements of the process, the reactor internals for all currently licensed and operating PWR designs in the U.S. were evaluated in the MRP Program and appropriate inspection, evaluation, and implementation requirements for reactor internals were defined.

Based on the completed evaluations, the RVI components are categorized within MRP-227-A as "Primary" components, "Expansion" components, "Existing Programs" components, or "No Additional Measures" components, described as follows:

- <u>Primary</u>

  Those PWR internals that are highly susceptible to the effects of at least one of the eight aging mechanisms were placed in the Primary group. The aging management requirements that are needed to ensure functionality of Primary components are described in the I&E guidelines. The Primary group also includes components that have shown a degree of tolerance to a specific aging

degradation effect, but for which no highly susceptible component exists or for which no highly susceptible component is accessible.

- Expansion

  Those PWR internals that are highly or moderately susceptible to the effects of at least one of the eight aging mechanisms, but for which functionality assessment has shown a degree of tolerance to those effects, were placed in the Expansion group. The schedule for implementation of aging management requirements for Expansion components depends on the findings from the examinations of the Primary components at individual plants.

- Existing Programs

  Those PWR internals that are susceptible to the effects of at least one of the eight aging mechanisms and for which generic and plant-specific existing AMP elements are capable of managing those effects, were placed in the Existing Programs group.

- No Additional Measures Programs

  Those PWR internals for which the effects of all eight aging mechanisms are below the screening criteria were placed in the No Additional Measures group. Additional components were placed in the No Additional Measures group as a result of a failure mode, effects, and criticality analysis (FMECA) and the functionality assessment. No further action is required by these guidelines for managing the aging of the No Additional Measures components.

The categorization and analysis used in the development of MRP-227-A are not intended to supersede any ASME B&PV Code Section XI [5] requirements. Any components that are classified as core support structures, as defined in ASME B&PV Code Section XI IWB 2500, Category B-N-3, have requirements that remain in effect and may only be altered as allowed by 10 CFR 50.55a.

4.3.1.2 NEI 03-08 Guidance within MRP-227-A

The industry program requirements of MRP-227-A are classified in accordance with the requirements of the NEI 03-08 protocols. The MRP-227-A guideline includes Mandatory and Needed elements as follows:

- **Mandatory**

There is one Mandatory element:

1. *Each commercial U.S. PWR unit shall develop and document a program for management of aging of reactor internals components within thirty-six months following issuance of MRP-227-Rev. 0 (that is, no later than December 31, 2011).*

   SGS Unit 1 Applicability: MRP-227, Revision 0 was officially issued by the industry in December 2008. An AMP must therefore be developed by December 2011. PSEG Nuclear

developed procedure ER-AP-333, "Pressurized Water Reactor Vessel Internals Program," [2] and Revision 0 of this AMP to meet the commitment that is contained in MRP-227, Revision 0.

According to the NRC RIS [4], SGS Unit 1 qualifies as a Category B plant because they have a renewed license with a commitment to submit an AMP/inspection plan based on MRP-227-A but that have not yet been required to do so by their commitment. This AMP fulfills the license renewal commitment to submit an implementation schedule for SGS Unit 1 in accordance with MRP-227-A [6] to the NRC no later than August 13, 2014.

- **Needed**

There are five Needed elements, with the fifth element being conditional based on examination results:

1. *Each commercial U.S. PWR unit shall implement MRP-227-A, Tables 4-1 through 4-9 and Tables 5-1 through 5-3 for the applicable design within twenty-four months following issuance of MRP-227-A.*

   SGS Unit 1 Applicability: MRP-227-A augmented inspections will be incorporated into the SGS ISI for the period of extended operation. The applicable Westinghouse tables contained in MRP-227-A include Table 4-3 (Primary), Table 4-6 (Expansion), Table 4-9 (Existing), and Table 5-3 (Examination Acceptance and Expansion Criteria) and are attached herein as Appendix C, Tables C-1, C-2, C-3, and C-4 respectively.

2. *Examinations specified in the MRP-227-A guidelines shall be conducted in accordance with the Inspection Standard, MRP-228* [15].

   SGS Unit 1 Applicability: Inspection standards will be in accordance with the requirements of MRP-228 [15]. These inspection standards will be used for augmented inspection at SGS Unit 1 as applicable where required by MRP-227-A directives.

3. *Examination results that do not meet the examination acceptance criteria defined in Section 5 of the MRP-227-A guidelines shall be recorded and entered in the plant corrective action program and dispositioned.*

   SGS Unit 1 Applicability: SGS Unit 1 will comply with this requirement.

4. *Each commercial U.S. PWR unit shall provide a summary report of all inspections and monitoring, items requiring evaluation, and new repairs to the MRP Program Manager within 120 days of the completion of an outage during which PWR internals within the scope of MRP-227-A are examined.*

   SGS Unit 1 Applicability: As discussed in Section 4.3.3, PSEG Nuclear will participate in future industry efforts and will adhere to industry directives for reporting, response, and follow-up.

5. *If an engineering evaluation is used to disposition an examination result that does not meet the examination acceptance criteria in Section 5, this engineering evaluation shall be conducted in accordance with a NRC-approved evaluation methodology.*

SGS Unit 1 Applicability: SGS Unit 1 will evaluate any examination results that do not meet the examination acceptance criteria in Section 5 of MRP-227-A in accordance with an NRC-approved methodology.

4.3.1.3 GALL AMP Development Guidance

It should be noted that Section XI.M16A of NUREG-1801, Revision 2 [18] includes a description of the attributes that make up an acceptable AMP. These attributes are consistent with the SGS Unit 1 AMR process. Evaluation of the SGS Unit 1 RVI AMP against GALL attribute elements is provided in Section 5 of this AMP.

As part of its license renewal, PSEG Nuclear is committed to participate in industry activities associated with the development of the standard Industry Guideline for Inspection and Evaluation of Reactor Internals. The industry efforts have defined the required inspections and examination techniques for those components critical to aging management of RVI. The results of the industry recommended inspections, as published in MRP-227-A, serve as the basis for identifying any augmented inspections that are required to complete the SGS Unit 1 RVI AMP.

4.3.1.4 MRP-227-A Applicability to SGS Unit 1

The applicability of MRP-227-A to SGS Unit 1 requires compliance with the following MRP-227-A assumptions:

- *30 years of operation with high-leakage core loading patterns (fresh fuel assemblies loaded in peripheral locations) followed by implementation of a low-leakage fuel management strategy for the remaining 30 years of operation.*

  SGS Unit 1 Applicability: SGS Unit 1 fuel management program changed from a high- to a low-leakage core-loading pattern prior to 30 years of operation [41].

- *Base load operation, i.e., typically operates at fixed power levels and does not usually vary power on a calendar or load demand schedule.*

  SGS Unit 1 Applicability: SGS Unit 1 operates as a base load unit [41].

- *No design changes beyond those identified in general industry guidance or recommended by the original vendors.*

  SGS Unit 1 Applicability: MRP-227-A states that the recommendations are applicable to all U.S. PWR operating plants as of May 2007 for the three designs identified. SGS Unit 1 has not made any modifications to reactor internals components beyond those identified in general industry guidance or recommended by the original vendor since May 2007 [41].

Based on the applicability, as stated above, the MRP-227-A work is representative for SGS Unit 1.

4.3.2 WCAP-17451-P, Reactor Internals Guide Tube Wear

The PWROG recently funded a program to develop a tool to facilitate prediction of continued operation of reactor upper internals guide tubes from a guide card and lower guide tube continuous guidance wear standpoint, as well as to establish an initial inspection schedule based on the various guide tube designs for the utilities participating in this program. A technical basis document was created for this program, WCAP-17451-P, Revision 1, "Reactor Internals Guide Tube Wear – Westinghouse Domestic Fleet Operational Projections" [44] which developed a guide plate (card) initial inspection schedule for Westinghouse NSSS designed plants. Per EPRI interim guidance letter, MRP 2014-006 [48], MRP group members have endorsed the guide plate (card) inspection requirements to be adopted within the next revision to MRP-227-A [6].

SGS Unit 1 is a four loop plant with a 17x17 standard guide tube design. According to Section 5.4 of the WCAP-17451-P [44], the generic initial guide card and continuous guidance inspection measurement effective full-power years (EFPY) range for this guide tube design is 24 to 28 EFPY. SGS Unit 1 was evaluated as a part of this technical basis and therefore, an alternative initial inspection measurement can be performed during an outage within a time range from 24 to 32 EFPY.

4.3.3 Ongoing Industry Programs

The U.S. industry, through both the EPRI/MRP and the PWROG, continues to sponsor activities related to RVI aging management, including planned development of a standard NRC submittal template, development of a plant-specific implementation program template for currently licensed U.S. PWR plants, and development of acceptance criteria and inspection disposition processes. PSEG Nuclear will maintain cognizance of industry activities related to PWR internals inspection and aging management; and will address/implement industry guidance stemming from those activities, as appropriate under NEI 03-08 practices.

4.4 SUMMARY

It should be noted that the PSEG Nuclear, the MRP, and the PWROG approaches to aging management are based on the GALL approach to aging management strategies. This approach includes a determination of which reactor internals passive components are most susceptible to the aging mechanisms of concern followed by determination of the proper inspection or mitigating program to provide reasonable assurance that the component will continue to perform its intended function through the period of extended operation. The GALL-based approach was used for the initial basis of the SGS LRA [1] that resulted in the NRC SER in NUREG-2101 [3].

The approach used to develop the SGS Unit 1 AMP is fully compliant with regulatory directives and approved documents. The additional evaluations and analysis completed by the MRP industry group have provided clarification on the level of inspection quality needed to determine the proper examination method and frequencies. The tables provided in MRP-227-A and included as Appendix C of this AMP provide the level of examination required for each of the components evaluated.

It is the PSEG Nuclear position that the SGS LRA, combined with any additional augmented inspections required by the MRP-227-A industry tables provided in Appendix C, provides reasonable assurance that

the reactor internals passive components will continue to perform their intended functions through the period of extended operation.

5 SALEM NUCLEAR GENERATING STATION REACTOR INTERNALS AGING MANAGEMENT PROGRAM ATTRIBUTES

The SGS Unit 1 RVI AMP is credited for aging management of RVI components for the following eight aging degradation mechanisms and their associated effects:

- Stress corrosion cracking
- Irradiation-assisted stress corrosion cracking
- Wear
- Fatigue
- Thermal aging embrittlement
- Irradiation embrittlement
- Void swelling and irradiation growth
- Thermal and irradiation-enhanced stress relaxation or irradiation-enhanced creep

The attributes of the SGS Unit 1 RVI AMP and compliance with NUREG-1801 (GALL Report), Section XI.M16A, "PWR Vessel Internals" [18] are described in this section. The GALL identifies 10 attributes for successful component aging management. The framework for assessing the effectiveness of the projected program is established by the use of the 10 elements of the GALL.

PSEG Nuclear fully utilized the GALL process contained in NUREG-1801, Revision 1 [42], in performing the AMR of the reactor internals in the license renewal process. However, PSEG Nuclear made a commitment [1, 3] to incorporate the following: (1) SGS Unit 1 will continue to participate in the industry programs for investigating and managing aging effects on reactor internals, (2) SGS Unit 1 will evaluate and implement the results of the industry programs as applicable to the reactor internals, and (3) upon completion of these programs, but not less than 24 months before entering the period of extended operation, SGS Unit 1 will submit an inspection plan for reactor internals to the NRC for review and approval. Augmented inspections, based on required program enhancements resulting from industry programs, will become part of the ASME B&PV Code, Section XI Program.

This AMP is consistent with that process, includes consideration of the augmented inspections identified in MRP-227-A, and fully meets the requirements of the commitment and GALL, Revision 2. Specific details of the SGS Unit 1 reactor internals AMP are summarized in the following subsections.

5.1 GALL REVISION 2 ELEMENT 1: SCOPE OF PROGRAM

GALL Report AMP Element Description

The scope of the program includes all RVI components at the SGS Unit 1 Nuclear Plant, which is built to a Westinghouse NSSS design. The scope of the program applies the methodology and guidance in the most recently NRC-endorsed version of MRP-227, which provides augmented inspection and flaw evaluation methodology for assuring the functional integrity of safety-related internals in commercial operating U.S. PWR nuclear power plants designed by B&W, CE, and Westinghouse. The scope of components considered for inspection under MRP-227 guidance includes core support structures (typically denoted as Examination Category B-N-3 by the ASME Code, Section XI), those RVI components that serve an intended license renewal safety function pursuant to criteria in 10 CFR 54.4(a)(1), and other RVI components whose failure could prevent

satisfactory accomplishment of any of the functions identified in 10 CFR 54.4(a)(1)(i), (ii), or (iii). The scope of the program does not include consumable items, such as fuel assemblies, reactivity control assemblies, and nuclear instrumentation, because these components are not typically within the scope of the components that are required to be subject to an aging management review (AMR), as defined by the criteria set in 10 CFR 54.21(a)(1). The scope of the program also does not include welded attachments to the internal surface of the reactor vessel because these components are considered to be ASME Code Class 1 appurtenances to the reactor vessel and are adequately managed in accordance with an applicant's AMP that corresponds to GALL AMP XI.M1, "ASME Code, Section XI Inservice Inspection, Subsections IWB, IWC, and IWD."

The scope of the program includes the response bases to applicable license renewal applicant action items (LRAAIs) on the MRP-227 methodology, and any additional programs, actions, or activities that are discussed in these LRAAI responses and credited for aging management of the applicant's RVI components. The LRAAIs are identified in the staff's safety evaluation on MRP-227 and include applicable action items on meeting those assumptions that formed the basis of the MRP's augmented inspection and flaw evaluation methodology (as discussed in Section 2.4 of MRP-227), and NSSS vendor-specific or plant-specific LRAAIs as well. The responses to the LRAAIs on MRP-227 are provided in Appendix C of the LRA.

The guidance of MRP-227 specifies applicability limitations to base-loaded plants and the fuel loading management assumptions upon which the functionality analyses were based. These limitations and assumptions require a determination of applicability by the applicant for each reactor and are covered in Section 2.4 of MRP-227 [18].

SGS Unit 1 Program Scope

The SGS Unit 1 reactor vessel internals consist of the upper core support structure, the lower core support structure, and the in-core instrumentation support structure, where each of these major components has a distinct purpose. Also included are the flux thimble tubes that extend from the penetrations on the reactor vessel lower head up to the seal table. The reactor vessel internals also include the fuel assemblies and the rod cluster control assemblies that are supported by all three structures. In addition, the major structural welds that form or join the major structures, the minor structural welds joining parts such as lifting lugs, supports, and tubes to the major structures, and the fasteners and alignment pins that guide, align, and fasten the major structures are within the scope of the reactor vessel internals. Additional reactor vessel internals details are provided in the SGS LRA [1].

The SGS Unit 1 reactor vessel internals subcomponents that require aging management review are indicated in Table 3.1.2-3 in the SGS LRA [1]. The table lists each subcomponent's intended function(s), material, and the aging effects that require management. A column in the tables lists the aging management program that is credited to address the component and aging effect during the period of extended operation. The NRC has reviewed and approved the aging management strategy presented in the Appendix B tables, as documented in the SER on license renewal [3].

The results of the industry research provided by MRP-227-A, summarized in the tables of Appendix C, provide the basis for the required augmented inspections, inspection techniques to permit detection and

characterizing of the aging effects (cracks, loss of material, loss of preload, etc.) of interest, prescribed frequency of inspection, and examination acceptance criteria. The information provided in MRP-227-A is rooted in the GALL methodology. The basic assumptions of MRP-227-A, Section 2.4 are met by SGS Unit 1 and are addressed in subsection 4.3.1.4 of this AMP. The Topical Report Conditions and Applicant/Licensee Action Items provided by the NRC in the Safety Evaluation (SE) on MRP-227, Revision 0 [6] are met by SGS and demonstration of compliance is addressed in Section 6.1 for the Topical Report Conditions and in Section 6.2 for the Applicant/Licensee Action Items. The SGS Unit 1 RVI AMP scope is additionally based on previously established and approved GALL Report approaches through application of the MRP-227-A [6] methodologies to determine those components that require aging management.

Conclusion

This element complies with the corresponding aging management attribute in NUREG-1801, Section XI.M16A [18] and Commitment 7 in the SER [3].

5.2 GALL REVISION 2 ELEMENT 2: PREVENTIVE ACTIONS

GALL Report AMP Element Description

> *The guidance in MRP-227 relies on PWR water chemistry control to prevent or mitigate aging effects that can be induced by corrosive aging mechanisms (e.g., loss of material induced by general, pitting corrosion, crevice corrosion, or stress corrosion cracking or any of its forms [SCC, PWSCC, or IASCC]). Reactor coolant water chemistry is monitored and maintained in accordance with the Water Chemistry Program. The program description, evaluation, and technical basis of water chemistry are presented in GALL AMP XI.M2, "Water Chemistry"* [18].

SGS Unit 1 Preventive Action

The SGS Unit 1 RVI AMP includes the Primary Water Chemistry Program [19, 20] as an existing program that complies with the requirements of this element. A description and applicability to the SGS Unit 1 RVI AMP is provided in the following subsection.

Primary Water Chemistry Program

To mitigate aging effects on component surfaces that are exposed to water as process fluid, chemistry programs are used to control water chemistry for impurities (e.g., dissolved oxygen, chloride, fluoride, and sulfate) that accelerate corrosion. This program relies on monitoring and control of water chemistry to keep peak levels of various contaminants below the system-specific limits. The SGS PWR Primary Water Chemistry Program [19, 20] is based on the current, approved revisions of EPRI PWR Primary Water Chemistry Guidelines [13].

This program is consistent with the corresponding program described in the GALL Report [18].

The limits of known detrimental contaminants imposed by the chemistry monitoring program are consistent with the EPRI PWR Primary Water Chemistry Guidelines [13].

Conclusion

This element complies with the corresponding aging management attribute in NUREG-1801, Section XI.M16A [18] and Commitment 7 in the SER [3].

5.3 GALL REVISION 2 ELEMENT 3: PARAMETERS MONITORED OR INSPECTED

GALL Report AMP Element Description

The program manages the following age-related degradation effects and mechanisms that are applicable in general to the RVI components at the facility: (a) cracking induced by SCC, PWSCC, IASCC, or fatigue/cyclical loading; (b) loss of material induced by wear; (c) loss of fracture toughness induced by either thermal aging or neutron irradiation embrittlement; (d) changes in dimension due to void swelling and irradiation growth, distortion, or deflection; and (e)loss of preload caused by thermal and irradiation-enhanced stress relaxation or creep. For the management of cracking, the program monitors for evidence of surface breaking linear discontinuities if a visual inspection technique is used as the non-destruction examination (NDE) method, or for relevant flaw presentation signals if a volumetric UT method is used as the NDE method. For the management of loss of material, the program monitors for gross or abnormal surface conditions that may be indicative of loss of material occurring in the components. For the management of loss of preload, the program monitors for gross surface conditions that may be indicative of loosening in applicable bolted, fastened, keyed, or pinned connections. The program does not directly monitor for loss of fracture toughness that is induced by thermal aging or neutron irradiation embrittlement, or by void swelling and irradiation growth; instead, the impact of loss of fracture toughness on component integrity is indirectly managed by using visual or volumetric examination techniques to monitor for cracking in the components and by applying applicable reduced fracture toughness properties in the flaw evaluations if cracking is detected in the components and is extensive enough to warrant a supplemental flaw growth or flaw tolerance evaluation under the MRP-227 guidance or ASME Code, Section XI requirements. The program uses physical measurements to monitor for any dimensional changes due to void swelling, irradiation growth, distortion, or deflection.

Specifically, the program implements the parameters monitored/inspected criteria for Westinghouse designed Primary Components in Table 4-3 of MRP-227. Additionally, the program implements the parameters monitored/inspected criteria for Westinghouse designed Expansion Components in Table 4-6 of MRP-227. The parameters monitored/inspected for Existing Program Components follow the bases for referenced Existing programs, such as the requirements for ASME Code Class RVI components in ASME Code, Section XI, Table IWB-2500-1, Examination Categories B-N-3, as implemented through the applicant's ASME Code, Section XI program, or the recommended program for inspecting Westinghouse-designed flux thimble tubes in GALL AMP XI.M37, "Flux Thimble Tube Inspection." No inspections, except for those specified in ASME Code, Section XI, are required for components that are identified as requiring "No Additional Measure," in accordance with the analyses reported in MRP-227 [18].

SGS Unit 1 Parameters Monitored or Inspected

The SGS Unit 1 AMP monitors, inspects, and/or tests for the effects of the eight aging degradation mechanisms on the intended function of the SGS Unit 1 PWR internals components through inspection and condition monitoring activities in accordance with the augmented requirements defined under industry directives as contained in MRP-227-A and ASME Section XI [5].

This AMP implements the requirements for the Primary Component inspections from Table 4-3 of MRP-227-A (included in Appendix C of this AMP as Table C-1), the Expansion Component inspections from Table 4-6 of MRP-227-A (included in Appendix C of this AMP as Table C-2), and the Existing Component inspections from Table 4-9 of MRP-227-A (included in Appendix C of this AMP as Table C-3). These tables contain requirements to monitor and inspect the RVI through the period of extended operation to address the effects of the eight aging degradation mechanisms. It is noted in Appendix C, Table C-1 that the PWROG has recently developed initial examination period requirements for guide plate (card) wear for Westinghouse NSSS designed plants [44] that will replace the current requirements in MRP-227-A [6].

For license renewal, the ASME Section XI Program [10] consists of periodic volumetric, surface, and/or visual examination of components for assessment, signs of degradation, and corrective actions. The requirements of MRP-227-A only augment and do not replace or modify the requirements of ASME Section XI. This program is consistent with the corresponding program described in the GALL Report [18].

Appendices B and C of this AMP provide a detailed listing of the components and subcomponents and the parameters monitored, inspected, and/or tested.

Conclusion

This element complies with or exceeds the corresponding aging management attribute in NUREG-1801, Section XI.M16A [18] and Commitment 7 in the SER [3].

5.4 GALL REVISION 2 ELEMENT 4: DETECTION OF AGING EFFECTS

GALL Report AMP Element Description

> *The detection of aging effects is covered in two places: (a) the guidance in Section 4 of MRP-227 provides an introductory discussion and justification of the examination methods selected for detecting the aging effects of interest; and (b) standards for examination methods, procedures, and personnel are provided in a companion document, MRP-228. In all cases, well-established methods were selected. These methods include volumetric UT examination methods for detecting flaws in bolting, physical measurements for detecting changes in dimension, and various visual (VT-3, VT-1, and EVT-1) examinations for detecting effects ranging from general conditions to detection and sizing of surface-breaking discontinuities. Surface examinations may also be used as an alternative to visual examinations for detection and sizing of surface-breaking discontinuities.*

Cracking caused by SCC, IASCC, and fatigue is monitored/inspected by either VT-1 or EVT-1 examination (for internals other than bolting) or by volumetric UT examination (bolting). The VT-3 visual methods may be applied for the detection of cracking only when the flaw tolerance of the component or affected assembly, as evaluated for reduced fracture toughness properties, is known and has been shown to be tolerant of easily detected large flaws, even under reduced fracture toughness conditions. In addition, VT-3 examinations are used to monitor/inspect for loss of material induced by wear and for general aging conditions, such as gross distortion caused by void swelling and irradiation growth or by gross effects of loss of preload caused by thermal and irradiation-enhanced stress relaxation and creep.

In addition, the program adopts the recommended guidance in MRP-227 for defining the Expansion criteria that needed to be applied to inspections of Primary Components and Existing Requirement Components and for expanding the examinations to include additional Expansion Components. As a result, inspections performed on the RVI components are performed consistent with the inspection frequency and sampling bases for Primary Components, Existing Requirement Components, and Expansion Components in MRP-227, which have been demonstrated to be in conformance with the inspection criteria, sampling basis criteria, and sample Expansion criteria in Section A.1.2.3.4 of NRC Branch Position RLSB-1.

Specifically, the program implements the parameters monitored/inspected criteria and bases for inspecting the relevant parameter conditions for Westinghouse designed Primary Components in Table 4-3 of MRP-227 and for Westinghouse designed Expansion Components in Table 4-6 of MRP-227.

The program is supplemented by the following plant-specific Primary Component and Expansion Component inspections for the program (as applicable): for SGS Unit 1, no additional Primary or Expansion components are relevant to the scope of aging management for the RVI.

In addition, in some cases (as defined in MRP-227), physical measurements are used as supplemental techniques to manage for the gross effects of wear, loss of preload due to stress relaxation, or for changes in dimension due to void swelling, deflection or distortion. The physical measurements methods applied in accordance with this program include that for the hold down spring. The hold down spring at SGS Unit 1 is fabricated from Type 304 SS that requires inspection by physical measurement [18].

SGS Unit 1 Detection of Aging Effects

Detection of indications that are required by the ASME Code Section XI ISI Program [10] is well established and field-proven through the application of the Section XI ISI Program. Those augmented inspections that are taken from the MRP-227-A recommendations will be applied through use of the MRP-228 inspection standard. This AMP implements the augmented inspection requirements of Table 4-3, Table 4-6, and Table 4-9 from MRP-227-A for the Primary, Expansion, and Existing Components, respectively. These are included in Appendix C of this AMP for reference. These tables include the inspection frequency and sampling basis. For the Expansion Components of MRP-227-A, this AMP implements the expansion requirements of Table 5-3 of MRP-227-A (included in Appendix C of this AMP as Table C-4).

Inspection can be used to detect physical effects of degradation including cracking, fracture, wear, and distortion. The choice of an inspection technique depends on the nature and extent of the expected damage. The recommendations supporting aging management for the reactor internals, as contained in this report, are built around three basic inspection techniques: (1) visual, (2) ultrasonic, and (3) physical measurement. The three different visual testing (VT) and enhanced visual testing (EVT) techniques are VT-3, VT-1, and EVT-1. The assumptions and process used to select the appropriate inspection technique are described in the following subsections. Inspection standards developed by the industry for the application of these techniques for augmented reactor internals inspections are documented in MRP-228 [15].

VT-1 Visual Examinations

The acceptance criteria for visual examinations conducted under categories B-N-2 (welded core support structures and interior attachments to reactor vessels) and B-N-3 (removable core support structures) are defined in IWB-3520 [5]. VT-1 visual examination is intended to identify crack-like surface flaws. Unacceptable conditions for a VT-1 examination are:

- Crack-like surface flaws on the welds joining the attachment to the vessel wall that exceed the allowable linear flaw standards of IWB-3510 [5]

- Structural degradation of attachment welds such that the original cross-sectional area is reduced by more than 10 percent

These requirements are defined to ensure the integrity of attachment welds on the ferritic pressure vessel. Although the IWB-3520 criteria do not directly apply to austenitic stainless steel internals, the clear intent is to ensure that the structure will meet minimum flaw tolerance fracture requirements. In the MRP-227-A recommendations, VT-1 examinations have been identified for components requiring close visual examinations with some estimate of the scale of deformation or wear. Note that in MRP-227-A, VT-1 has only been selected to detect distortion as evidenced by small gaps between the upper-to-lower mating surfaces of CE-welded core shrouds assembled in two vertical sections. Therefore, no additional VT-1 inspections over and above those required by ASME Section XI ISI have been specified.

EVT-1 Enhanced Visual Examination for the Detection of Surface Breaking Flaws

In the augmented inspections detailed in MRP-227-A for reactor internals, the EVT-1 enhanced visual examination has been identified for inspection of components where surface-breaking flaws are a potential concern. Any visual inspection for cracking requires a reasonable expectation that the flaw length and crack mouth opening displacement meet the resolution requirements of the observation technique. The EVT-1 specification augments the VT-1 requirements to provide more rigorous inspection standards for stress corrosion cracking and has been demonstrated for similar inspections in boiling water reactor (BWR) internals. Enhanced visual examination (i.e., EVT-1) is also conducted in accordance with the requirements described for visual examination (i.e., VT-1) with additional requirements (such as camera scanning speed) currently being developed by the industry. Any recommendation for EVT-1 inspection will require additional analysis to establish flaw-tolerance criteria, which must take into account potential embrittlement due to thermal aging or neutron irradiation. The industry through the PWROG has developed an approach for acceptance criteria methodologies to

support plant-specific augmented examinations. This work is summarized in WCAP-17096-NP "Reactor Internals Acceptance Criteria Methodology and Data Requirements" [16]. The acceptance criteria developed using these methodologies may be created on either a generic or plant-specific basis because both loads and component dimensions may vary from plant to plant within a typical PWR design.

VT-3 Examination for General Condition Monitoring

In the augmented inspections detailed in the MRP-227-A for reactor internals, the VT-3 visual examination has been identified for inspection of components where general condition monitoring is required. The VT-3 examination is intended to identify individual components with significant levels of existing degradation. As the VT-3 examination is not intended to detect the early stages of component cracking or other incipient degradation effects, it should not be used when failure of an individual component could threaten either plant safety or operational stability. The VT-3 examination may be appropriate for inspecting highly redundant components (such as baffle-edge bolts), where a single failure does not compromise the function or integrity of the critical assembly.

The acceptance criteria for visual examinations conducted under categories B-N-2 (welded core support structures and interior attachments to reactor vessels) and B-N-3 (removable core support structures) are defined in IWB-3520. These criteria are designed to provide general guidelines. The unacceptable conditions for a VT-3 examination are:

- Structural distortion or displacement of parts to the extent that component function may be impaired;
- Loose, missing, cracked, or fractured parts, bolting, or fasteners;
- Foreign materials or accumulation of corrosion products that could interfere with control rod motion or could result in blockage of coolant flow through fuel;
- Corrosion or erosion that reduces the nominal section thickness by more than 5 percent;
- Wear of mating surfaces that may lead to loss of function;
- Structural degradation of interior attachments such that the original cross-sectional area is reduced more than 5 percent.

The VT-3 examination is intended for use in situations where the degradation is readily observable. It is meant to provide an indication of condition, and quantitative acceptance criteria are not generally required. In any particular recommendation for VT-3 visual examination, it should be possible to identify the specific conditions of concern. For instance, the unacceptable conditions for wear indicate wear that might lead to loss of function. Guidelines for wear in a critical-alignment component may be very different from the guidelines for wear in a large structural component.

Surface Examination

In order to further characterize discontinuities on the surface of components, surface examination can supplement either visual (VT-3) or (VT-1/EVT-1) examinations specified in the MRP-227-A guidelines. This supplemental examination may thus be used to reject or accept relevant indications. A surface examination is an examination that indicates the presence of surface discontinuities, and the ASME B&PV Code [5] lists magnetic particle, liquid penetrant, eddy current, and ultrasonic examination methods as surface examination alternatives. Here, only the electromagnetic testing (ET), also called eddy current surface examination method, is covered.

When selected for use as a supplemental examination to examinations performed in the MRP-227-A guidelines, an ET examination is conducted in accordance with the requirements of the inspection standard [15].

ET examination is widely used for heat exchanger tubing inspections. Eddy currents are induced in the inspected object by electromagnetic coils, with disruptions in the eddy current flow caused by surface or near-surface anomalies detected by suitable instrumentation. Industry experience with ET examination is relatively robust, especially in the aerospace and petroleum refinery industries. The experience base for PWR nuclear systems is moderately robust, in particular for examination of steam generator, flux thimble, and heat exchanger tubing.

Ultrasonic Testing

Volumetric examinations in the form of ultrasonic testing (UT) techniques can be used to identify and determine the length and depth of a crack in a component. Although access to the surface of the component is required to apply the ultrasonic signals, the flaw may exist in the bulk of the material. In this proposed strategy, UT inspections have been recommended exclusively for detection of flaws in bolts. For the bolt inspections, any bolt with a detected flaw should be assumed to have failed. The size of the flaw in the bolt is not critical because crack growth rates are generally high, and it is assumed that the observed flaw will result in failure prior to the next inspection opportunity. It has generally been observed through examination performance demonstrations that UT can reliably (90 percent or greater reliability) detect flaws that reduce the cross-sectional area of a bolt by 35 percent.

Failure of a single bolt does not compromise the function of the entire assembly. Bolting systems in the reactor internals are highly redundant. For any system of bolts, it is possible to demonstrate multiple acceptable bolting patterns. The evaluation program must demonstrate that the remaining bolts meet the requirements for an acceptable bolting pattern for continued operation. The evaluation procedures must also demonstrate that the pattern of remaining bolts contains sufficient margin such that continuation of the bolt failure rate will not result in failure of the system to meet the requirements for an acceptable bolting pattern before the next inspection.

Establishment of the acceptable bolting pattern for any system of bolts requires analysis to demonstrate that the system will maintain reliability and integrity in continuing to perform the intended function of the component. This analysis is highly plant-specific. Therefore, any recommendation for UT inspection of bolts assumes that the plant owner will work with the designer to establish acceptable bolting patterns prior to the inspection to support continued operation.

Physical Measurement Examination

Continued functionality can be confirmed by physical measurements to evaluate the impact caused by various degradation mechanisms such as wear or loss of functionality as a result of loss of preload or material deformation. For SGS Unit 1, direct physical measurements are required only for the hold down spring.

Conclusion

This element complies with or exceeds the corresponding aging management attribute in NUREG-1801, Section XI.M16A [18] and Commitment 7 in the SER [3].

5.5 GALL REVISION 2 ELEMENT 5: MONITORING AND TRENDING

GALL Report AMP Element Description

> *The methods for monitoring, recording, evaluating, and trending the data that result from the program's inspections are given in Section 6 of MRP-227 and its subsections. The evaluation methods include recommendations for flaw depth sizing and for crack growth determinations as well for performing applicable limit load, linear elastic and elastic-plastic fracture analyses of relevant flaw indications. The examinations and re-examinations required by the MRP-227 guidance, together with the requirements specified in MRP-228 for inspection methodologies, inspection procedures, and inspection personnel, provide timely detection, reporting, and corrective actions with respect to the effects of the age-related degradation mechanisms within the scope of the program. The extent of the examinations, beginning with the sample of susceptible PWR internals component locations identified as Primary Component locations, with the potential for inclusion of Expansion Component locations if the effects are greater than anticipated, plus the continuation of the Existing Programs activities, such as the ASME Code, Section XI, Examination Category B-N-3 examinations for core support structures, provides a high degree of confidence in the total program* [18].

SGS Unit 1 Monitoring and Trending

Operating experience with PWR reactor internals has been generally proactive. Flux thimble wear and control rod guide tube support pin cracking issues were identified by the industry and continue to be actively managed. The extremely low frequency of failure in reactor internals makes monitoring and trending based on OE somewhat impractical. The majority of the materials aging degradation models used to develop the MRP-227-A guidelines are based on test data from reactor internals components removed from service. The data is used to identify trends in materials degradation and forecast potential component degradation. The industry continues to share both material test data and OE through the auspices of the MRP and PWROG. PSEG Nuclear has in the past and will continue to maintain cognizance of industry activities and shared information related to PWR internals inspection and aging management as demonstrated in their quality programs [23, 24, 25, 26, 27, 28, 29, 30].

Nickel alloy reactor vessel internals components are included in the SGS PWR Vessel Internals program. The ASME Section XI ISI, Subsections IWB, IWC, and IWD, is used for monitoring and trending activities for loss of material and wear of nickel alloy reactor vessel internals.

Inspections credited in the SGS LRA are based on utilizing the 10-year ISI Program and the augmented inspections derived from MRP-227-A and repeated here in Appendix C. The MRP-227-A inspections only augment and do not replace the existing ASME Section XI ISI requirements. These inspections, where practical, are scheduled to be conducted in conjunction with typical 10-year ISI examinations.

Appendix C, Tables C-1, C-2, and C-3 identify the augmented Primary and Expansion inspection and monitoring recommendations, and the Existing programs credited for inspection and aging management. As discussed in MRP-227-A, inspection of the "Primary" components provides reasonable assurance for demonstrating component current capacity to perform the intended functions. It is noted in Appendix C, Table C-1 that the PWROG has recently developed initial examination period requirements for guide plate (card) wear for Westinghouse NSSS designed plants [44] that will replace the current requirements in MRP-227-A [6]. Table C-4 in Appendix C identifies the MRP-227-A expansion criteria from the Primary components. If these expansion criteria are met for a component, the associated Expansion component is to be inspected to manage the aging degradation.

Reporting requirements are included as part of the MRP-227-A guidelines. Consistent reporting of inspection results across all PWR designs will enable the industry to monitor reactor internals degradation on an ongoing industry basis as the period of extended operation moves forward. Reporting of examination results will allow the industry to monitor and trend results and take appropriate preemptive action through update of the MRP guidelines.

Conclusion

This element complies with or exceeds the corresponding aging management attribute in NUREG-1801, Section XI.M16A [18] and Commitment 7 in the SER [3].

5.6 GALL REVISION 2 ELEMENT 6: ACCEPTANCE CRITERIA

GALL Report AMP Element Description

Section 5 of MRP-227 provides specific examination acceptance criteria for the Primary and Expansion Component examinations. For components addressed by examinations referenced to ASME Code, Section XI, the IWB-3500 acceptance criteria apply. For other components covered by Existing Programs, the examination acceptance criteria are described within the Existing Program reference document.

The guidance in MRP-227 contains three types of examination acceptance criteria:

- *For visual examination (and surface examination as an alternative to visual examination), the examination acceptance criterion is the absence of any of the specific, descriptive relevant conditions; in addition, there are requirements to record and disposition surface breaking indications that are detected and sized for length by VT-1/EVT-1 examinations;*

- *For volumetric examination, the examination acceptance criterion is the capability for reliable detection of indications in bolting, as demonstrated in the examination Technical Justification; in addition, there are requirements for system-level assessment of bolted or pinned assemblies with unacceptable volumetric (UT) examination indications that exceed specified limits; and*

- *For physical measurements, the examination acceptance criterion for the acceptable tolerance in the measured differential height from the top of the plenum rib pads to the vessel seating surface in B&W plants are given in Table 5-1 of MRP-227. The acceptance criterion for physical measurements performed on the height limits of the Westinghouse-designed hold down springs are required for 304 SS hold down springs. SGS Unit 1 has a 304 SS hold down spring; therefore, SGS Unit 1 is required to produce acceptance criteria for the physical measurements on the hold down spring* [18].

SGS Unit 1 Acceptance Criteria

Those recordable indications that are the result of inspections required by the existing SGS ISI Program scope are evaluated in accordance with the applicable requirements of the ASME Code through the existing Corrective Action Program [23], inspection program [10], and repair programs [31].

Inspection acceptance and expansion criteria are provided in Appendix C, Table C-4. These criteria will be reviewed periodically as the industry continues to develop and refine the information and will be included in updates to SGS procedures to enable the examiner to identify examination acceptance criteria considering state-of-the-art information and techniques.

Augmented inspections, as defined by the MRP-227-A requirements included in this AMP as Appendix C, Table C-1, Table C-2, and Table C-3, that result in recordable relevant conditions will be entered into the plant Corrective Action Program and addressed by appropriate actions that may include enhanced inspection, repair, replacement, mitigation actions, or analytical evaluations. An example of an analytical evaluation is using an acceptable bolting WCAP approach [32], such as those commonly used to support continued component or assembly functionality. Additional analysis to establish acceptable bolting pattern evaluation criteria for the baffle-former bolt assembly, as contained in various industry documents [32], is also considered in determining the acceptance of inspection results to support continued component or assembly functionality.

The industry, through various cooperative efforts, is working to construct a consensus set of tools in line with accepted and proven methodologies to support this element. One of these tools is the PWROG document WCAP-17096-NP, "Reactor Internals Acceptance Criteria Methodology and Data Requirements," [16], which details acceptance criteria methodology for the MRP-227 Primary and Expansion components. Status is monitored through direct PSEG Nuclear cognizance of industry (including PWROG) activities related to PWR internals inspection and aging management.

Conclusion

This element complies with or exceeds the corresponding aging management attribute in NUREG-1801, Section XI.M16A [18] and Commitment 7 in the SER [3].

5.7 GALL REVISION 2 ELEMENT 7: CORRECTIVE ACTIONS

GALL Report AMP Element Description

Corrective actions following the detection of unacceptable conditions are fundamentally provided for in each plant's corrective action program. Any detected conditions that do not satisfy the examination acceptance criteria are required to be dispositioned through the plant corrective action program, which may require repair, replacement, or analytical evaluation for continued service until the next inspection. The disposition will ensure that design basis functions of the reactor internals components will continue to be fulfilled for all licensing basis loads and events. Examples of methodologies that can be used to analytically disposition unacceptable conditions are found in the ASME Code, Section XI or in Section 6 of MRP-227. Section 6 of MRP-227 describes the options that are available for disposition of detected conditions that exceed the examination acceptance criteria of Section 5 of the report. These include engineering evaluation methods, as well as supplementary examinations to further characterize the detected condition, or the alternative of component repair and replacement procedures. The latter are subject to the requirements of the ASME Code, Section XI. The implementation of the guidance in MRP-227, plus the implementation of any ASME Code requirements, provides an acceptable level of aging management of safety-related components addressed in accordance with the corrective actions of 10 CFR Part 50, Appendix B or its equivalent, as applicable.

Other alternative corrective action bases may be used to disposition relevant conditions if they have been previously approved or endorsed by the NRC. Examples of previously NRC-endorsed alternative corrective actions bases include those corrective actions bases for Westinghouse-design RVI components that are defined in Tables 4-1, 4-2, 4-3, 4-4, 4-5, 4-6, 4-7 and 4-8 of Westinghouse Report No. WCAP-14577-Rev. 1-A, or for B&W-designed RVI components in B&W Report No. BAW-2248. Westinghouse Report No. WCAP-14577-Rev. 1-A was endorsed for use in an NRC SE to the Westinghouse Owners Group, dated February 10, 2001. B&W Report No. BAW-2248 was endorsed for use in an SE to Framatome Technologies on behalf of the B&W Owners Group, dated December 9, 1999. Alternative corrective action bases not approved or endorsed by the NRC will be submitted for NRC approval prior to their implementation [18].

SGS Unit 1 Corrective Action

The existing SGS Unit 1 procedure for Inservice Repair and Replacement [10, 31] and the established 10 CFR 50, Appendix B, Program [23, 33, 34] that addresses the elements of corrective actions, confirmation process, and administrative controls will be credited for this element. The Inservice Repair and Replacement procedure establishes the SGS Unit 1 repair and replacement requirements of ASME Code Section XI, "Rules for ISI of Nuclear Power Plant Components" [5]. These requirements include the identification of a repair cycle, repair plan, and verification of acceptability for replacements. SGS Unit 1 is committed to developing corrective actions for augmented inspections using repair and replacement procedures equivalent to those requirements in ASME B&PV Code, Section XI and MRP-227-A, Section 6 [6].

Conclusion

This element complies with the corresponding aging management attribute in NUREG-1801, Section XI.M16A [18] and Commitment 7 in the SER [3].

5.8 GALL REVISION 2 ELEMENT 8: CONFIRMATION PROCESS

GALL Report AMP Element Description

> *Site quality assurance procedures, review and approval processes, and administrative controls are implemented in accordance with the requirements of 10 CFR Part 50, Appendix B, or their equivalent, as applicable. It is expected that the implementation of the guidance in MRP-227 will provide an acceptable level of quality for inspection, flaw evaluation, and other elements of aging management of the PWR internals that are addressed in accordance with the 10 CFR Part 50, Appendix B, or their equivalent (as applicable), confirmation process, and administrative controls* [18].

SGS Unit 1 Confirmation Process

SGS has an established 10 CFR 50, Appendix B, Program [23, 33, 34] that addresses the elements of corrective actions, confirmation process, and administrative controls. The SGS Unit 1 Program includes non-safety-related structures, systems, and components. Quality assurance (QA) procedures, review and approval processes, and administrative controls are implemented in accordance with the requirements of 10 CFR 50, Appendix B.

Conclusion

This element complies with or exceeds the corresponding aging management attribute in NUREG-1801, Section XI.M16A [18] and Commitment 7 in the SER [3].

5.9 GALL REVISION 2 ELEMENT 9: ADMINISTRATIVE CONTROLS

GALL Report AMP Element Description

> *The administrative controls for such programs, including their implementing procedures and review and approval processes, are under existing site 10 CFR 50 Appendix B Quality Assurance Programs, or their equivalent, as applicable. Such a program is thus expected to be established with a sufficient level of documentation and administrative controls to ensure effective long-term implementation* [18].

SGS Unit 1 Administrative Controls

SGS Unit 1 has an established 10 CFR Part 50, Appendix B Program [23, 33, 34] that addresses the elements of corrective actions, confirmation process, and administrative controls. The SGS Unit 1 Program includes non-safety-related structures, systems, and components. QA procedures, review and

approval processes, and administrative controls are implemented in accordance with the requirements of 10 CFR 50, Appendix B.

Conclusion

This element complies with or exceeds the corresponding aging management attribute in NUREG-1801, Section XI.M16A [18] and Commitment 7 in the SER [3].

5.10 GALL REVISION 2 ELEMENT 10: OPERATING EXPERIENCE

GALL Report AMP Element Description

> *Relatively few incidents of PWR internals aging degradation have been reported in operating U.S. commercial PWR plants. A summary of observations to date is provided in Appendix A of MRP-227-A. The applicant is expected to review subsequent operating experience for impact on its program or to participate in industry initiatives that perform this function.*
>
> *The application of the MRP-227 guidance will establish a considerable amount of operating experience over the next few years. Section 7 of MRP-227 describes the reporting requirements for these applications, and the plan for evaluating the accumulated additional operating experience* [18].

SGS Unit 1 Operating Experience

Extensive industry and SGS Unit 1 OE has been reviewed during the development of the RVI AMP. The experience reviewed includes NRC Information Notices 84-18, "Stress Corrosion Cracking in PWR Systems" [35] and 98-11, "Cracking of Reactor Vessel Internal Baffle Former Bolts in Foreign Plants" [36]. Most of the industry operating experience reviewed has involved cracking of austenitic stainless steel baffle-former bolts, or SCC of high-strength internals bolting. SCC of control rod guide tube support pins has also been reported.

Early plant OE related to hot functional testing and reactor internals is documented in plant historical records. Inspections, performed as part of the 10-year ISI Program, have been conducted as designated by existing commitments and would be expected to discover overall general internals structure degradation. To date, very little degradation has been observed industry-wide.

Industry OE is routinely reviewed by PSEG Nuclear engineers using Institute of Nuclear Power Operations (INPO) OE, the Nuclear Network, and other information sources as directed under the SGS operating experience procedure [37], for the determination of additional actions and lessons learned. These insights, as applicable, can be incorporated into the plant systems quarterly health reports and further evaluated for incorporation into plant programs.

A review of industry and plant-specific experience with RVI reveals that the U.S. industry, including PSEG Nuclear and SGS Unit 1, has responded proactively to industry issues relative to reactor internals degradation. An example that demonstrates this proactive response is the replacement of control rod guide tube support pins at SGS Unit 1. Other relevant operating experience includes the SGS Unit 1 flux

thimble tube replacement and the experience gained through the conduct of ASME Section XI ISI. These are briefly described in the following paragraphs.

- SGS Unit 1 Control Rod Guide Tubes Support Pin Replacement

  The control rod guide tube support pins were replaced at SGS Unit 1 during the S1R18 (2007) refueling outage (RO) [38]. The replacement pins included a material upgrade from X-750 to Type 316 stainless steel in support of managing aging in the component.

- SGS Unit 1 Flux Thimble Tubes Replacement

  As discussed in the SGS LRA [1], SGS Unit 1 replaced all of the flux thimble tubes during the 1RF7 (1990) outage with an improved design. The 1RF10 (1993) activities for SGS Unit 1 involved ECT of eleven of the improved design flux thimble tubes. The results indicated that there was no significant wear on any of the eleven inspected flux thimble tubes. Also, the examinations indicated that there was no cladding bulging or ovality detected.

- SGS Unit 1 ASME Section XI ISI, Subsection IWB, IWC, and IWD

  The operating experience of the ASME Section XI ISI, subsections IWB, IWC, and IWD program did not show any adverse trends in performance. Three specific examples of visual, ultrasonic, and dye penetrant examinations with indications [1] provide objective evidence that the effects of aging are effectively managed by the ASME Section XI ISI, subsections IWB, IWC, and IWD program

A key element of the MRP-227-A guideline is the reporting of age-related degradation of RVI components. PSEG Nuclear, through its participation in PWROG and EPRI-MRP activities, will continue to benefit from the reporting of inspection information and will share its own OE with the industry through the reporting requirements of Section 7 of MRP-227-A. The collected information from MRP-227-A augmented inspections will benefit the industry in its continued response to RVI aging degradation.

Conclusion

This element complies with or exceeds the corresponding aging management attribute in NUREG-1801, Section XI.M16A [18] and Commitment 7 in the SER [3].

6 DEMONSTRATION

SGS has demonstrated a long-term commitment to aging management of reactor internals. This AMP is based on an established history of programs to identify and monitor potential aging degradation in the reactor internals. Programs and activities undertaken in the course of fulfilling that commitment include:

- The examinations required by ASME Section XI for the SGS Unit 1 reactor vessel internals have been performed during each 10-year interval since plant operations commenced.
- As documented in SGS Unit 1 operational procedures, Operating Experience Reports (OER) are continuously reviewed by SGS Unit 1 personnel for applicable issues that indicate a need for updated operating procedures or programs.
- Review of Nuclear Oversight Section (NOS) audit reports, NRC inspection reports, and INPO evaluations indicate no unacceptable issues related to RVI inspections [23, 33, 34].
- The Primary Water Chemistry Program at SGS Unit 1 has been effective in maintaining the levels of oxygen, halides, and sulfate sufficiently low to prevent SCC of the reactor vessel internals
- Replacement control rod guide tube support pins for SGS Unit 1 in 2007 were fabricated from Type 316 stainless steel (SS), which will provide additional resistance to PWSCC.
- SGS Unit 1 replaced all of the flux thimble tubes during the 1RF7 (1990) outage with an improved design.
- SGS has participated in the PWROG program to develop initial examination period requirements for guide plate (card) wear for Westinghouse NSSS designed plants [44].
- PSEG Nuclear has actively participated in past and ongoing EPRI and PWROG RVI activities. PSEG Nuclear will continue to maintain cognizance of industry activities related to PWR internals inspection and aging management and will address/implement the industry guidance, stemming from those activities as appropriate under NEI 03-08 practices.

This AMP fulfills the approved license renewal methodology requirement to identify the most susceptible components and to inspect those components with an indication detection level commensurate with the expected degradation mechanism indication. Augmented inspections derived from the information contained in MRP-227-A, the industry I&E Guidelines, have been utilized in this AMP to build on existing plant programs. This approach is expected to encourage detection of a degradation mechanism at its first appearance, which is consistent with the ASME Code approach to inspections. This approach provides reasonable assurance that the internals components will continue to perform their intended function through the period of extended operation.

Typical ASME Code Section XI examinations identified in the AMP for the period of extended operation are currently scheduled to be performed at SGS Unit 1 during the 1RF26 (2019). The augmented inspections discussed in compliance with MRP-227-A requirements have been integrated in the implementation schedule, which is shown in Section 7. Integration of the required inspections will be

tracked to completion. As discussed, the industry MRP-227-A guidelines also provide for updates as experience is gained through inspection results. This feedback loop will enable updates based on actual inspection experience.

The augmented inspections described in this document, as summarized in Appendix C, combined with the ASME Code Section XI ISI Program inspections, existing SGS Programs, and use of OERs, provide reasonable assurance that the reactor internals at SGS Unit 1 will continue to perform their intended functions through the period of extended operation.

Table 6-1 lists the seven topical report conditions and Section 6.2 lists the eight applicant action items that came out of the NRC review of MRP-227, as listed in [6], as well as their compliance within this AMP.

6.1 DEMONSTRATION OF TOPICAL REPORT CONDITIONS COMPLIANCE TO SE ON MRP-227, REVISION 0

Table 6-1 Topical Report Condition Compliance to SE on MRP-227

Topical Condition	Applicable/Not Applicable	Compliance in AMP
1. High consequence components in the "No Additional Measures" Inspection Category	Applicable	The upper core plate and the lower support forging or casting components are added to Table C-2 as "Expansion Components" linked to the "Primary Component," the control rod guide tube (CRGT) lower flange weld.
2. Inspection of components subject to irradiation-assisted stress corrosion cracking	Applicable	The upper and lower core barrel cylinder girth welds and the lower core barrel flange weld are moved from Table C-2 "Expansion Components" to Table C-1 "Primary Components."
3. Inspection of high consequence components subject to multiple degradation mechanisms	Not Applicable	Not applicable since SGS Unit 1 is a Westinghouse designed reactor.
4. Imposition of minimum examination coverage criteria for "Expansion" inspection category components	Applicable	Notes 2 through 4 were added to Table C-1, as well as Note 2 to Table C-2 to reflect this condition.
5. Examination frequencies for baffle-former bolts and core shroud bolts	Applicable	Not applicable for the core shroud bolts since SGS Unit 1 is a Westinghouse designed reactor. In Table C-1 for the baffle-former bolts, the inspection frequency was changed to be 10 years following the initial or baseline inspection, unless SGS Unit 1 provides an evaluation for NRC staff approval that justifies a longer interval between inspections. A note has been added in Table C-1 to address the inspection frequency for the baffle-former bolts.

Table 6-1 Topical Report Condition Compliance to SE on MRP-227

Topical Condition	Applicable/Not Applicable	Compliance in AMP
6. Periodicity of the re-examination of "Expansion" inspection category components	Applicable	The re-inspection frequency is 10 years following initial inspection, unless SGS Unit 1 provides an evaluation for NRC staff approval that justifies a longer interval between inspections. A note has been added in Table C-2 to address the inspection frequency for the expansion components.
7. Updating of MRP-227, Revision 0, Appendix A	Applicable	Section 5 is updated to reflect XI.M16A from GALL Revision 2 [18].

6.2 DEMONSTRATION OF APPLICANT/LICENSEE ACTION ITEM COMPLIANCE TO SE ON MRP-227, REVISION 0

6.2.1 SE Applicant/Licensee Action Item 1: Applicability of FMECA and Functionality Analysis Assumptions

As addressed in Section 3.2.5.1 of this SE, each applicant/licensee is responsible for assessing its plant's design and operating history and demonstrating that the approved version of MRP-227 is applicable to the facility. Each applicant/licensee shall refer, in particular, to the assumptions regarding plant design and operating history made in the FMECA and functionality analyses for reactors of their design (i.e., Westinghouse, CE, or B&W) which support MRP-227 and describe the process used for determining plant-specific differences in the design of their RVI components or plant operating conditions, which result in different component inspection categories. The applicant/licensee shall submit this evaluation for NRC review and approval as part of its application to implement the approved version of MRP-227. This is Applicant/Licensee Action Item 1 [6].

SGS Unit 1 Compliance

The process used to verify that SGS Unit 1 is reasonably represented by the generic industry program assumptions with regard to neutron fluence, temperature, materials, and stress values used in the development of MRP-227-A is as follows:

1. Identification of typical Westinghouse pressurized water reactor (PWR) internals components (MRP-191, Table 4-4 [14]).

2. Identification of SGS Unit 1 PWR internals components.

3. Comparison of the typical Westinghouse PWR internals components to the SGS Unit 1 PWR internals components.

    a. Confirmation that no additional items were identified by this comparison (primarily supports Applicant/Licensee Action Item (A/LAI) 2).

b. Confirmation that the materials identified for SGS Unit 1 are consistent with those materials identified in MRP-191, Table 4-4.

c. Confirmation that the SGS Unit 1 internals are the same as, or equivalent to, the typical Westinghouse PWR internals regarding design and fabrication.

4. Confirmation that the SGS Unit 1 operating history is consistent with the assumptions in MRP-227-A regarding core loading patterns.

5. Confirmation that the SGS Unit 1 RVI materials operated at temperatures within the original design basis parameters.

6. Determination of stress values based on design basis documents.

7. Confirmation that any changes to the SGS Unit 1 RVI components do not impact the application of the MRP-227-A generic aging management strategy.

SGS Unit 1 reactor internals components are reasonably represented by the design and operating history assumptions regarding neutron fluence, temperature, materials, and stress values in the MRP-191 generic FMECA and the MRP-232 functionality analyses based on the following.

1. SGS Unit 1 operating history is consistent with the assumptions in MRP-227-A with regard to neutron fluence, as stated in Section 4.3.1.4.

   a. The FMECA and functionality analyses for MRP-227-A were based on the assumption of 30 years of operation with high-leakage core loading patterns followed by 30 years of low-leakage core fuel management strategy. In Fuel Cycle C13 (1998) [41] at approximately 21 years of operation, SGS Unit 1 switched to use of a low-leakage core design. Therefore, SGS Unit 1 meets the fluence and fuel management assumptions in MRP-191 and requirements for MRP-227-A application.

   b. SGS Unit 1 has operated under base load conditions over the life of the plant [41]. Therefore, SGS Unit 1 satisfies the assumptions in MRP documents regarding operational parameters affecting fluence.

2. The SGS Unit 1 RVI operate between T_{hot} and T_{cold} [41], which are not less than approximately 530.2°F for T_{cold} and not higher than 613.1°F for T_{hot}. The design temperature for the reactor vessel is 650°F. SGS Unit 1 operating history is within original design basis parameters and therefore consistent with the assumptions used to develop the MRP-227-A aging management strategy with regard to temperature operational parameters.

3. SGS Unit 1 internals components and materials are comparable to the typical Westinghouse PWR internals components (MRP-191, Table 4-4).

   a. Some of the components listed in the AMR are not contained in SGS Unit 1. PSEG was being all-encompassing and conservative in creating the AMR by including

components that were clearly a part of the SGS Unit 1 reactor internals and components which were not confirmed as part of the internals [41]. The work done for responding to A/LAIs 1 and 2 confirmed the actual list of components and identified those that were not a part of the SGS Unit 1 reactor internals. This review identified one component in the AMR [1], listed as the Upper Internals Assembly: Capped Top Thermocouple Columns, which is not included in MRP-191. However, the review also determined that this component is not applicable to SGS Unit 1. Therefore, SGS meets the requirements for aging as outlined in MRP-227-A.

b. Materials identified for SGS Unit 1 are consistent or nearly equivalent with those materials identified in MRP-191, Table 4-4 for Westinghouse-designed plants [14], except for the guide plates/cards, and brackets, clamps, terminal blocks, and conduit straps (conduit support, conduit support gusset, gusset clamp, and thermocouple stop) which are identified as being or having the potential to be cast CF8. However, based on a review by an expert panel [45], it was determined that this has no effect on the recommended MRP aging management inspection sampling strategy. The guide cards/plates will remain classified as a "Primary" inspection item, and the brackets, clamps, terminal blocks and conduit straps will remain classified as "No Additional Measures" items based on a consideration of the likelihood of failure and likelihood of damage. There is no change in MRP-227-A inspection requirements as a result of the inclusion of CF8 for these components. In addition, several other components have different materials than specified in MRP-191, but these material differences have been determined to have no effect of the recommended MRP aging management inspection sampling strategy [45].

c. SGS Unit 1 internals are the same as, or equivalent to, the typical Westinghouse PWR internals regarding design and fabrication.

4. Modifications to the SGS Unit 1 reactor internals made over the lifetime of the plant are those specifically directed by Westinghouse, the Original Equipment Manufacturer (OEM) [41]. The design has been maintained over the lifetime of the plant as specified by the OEM, operational parameters are compliant with MRP-227-A requirements with regard to fluence and temperature, and the components and materials are the same as, or nearly equivalent to, those considered in MRP-191. Therefore, the SGS Unit 1 stress values are represented by the assumptions in MRP-191, MRP-232, and MRP-227-A, confirming the applicability of the generic FMECA.

Conclusion

The assumptions regarding operating history made in the FMECA and functionality analyses for the Westinghouse design apply to SGS Unit 1. There are no components at SGS Unit 1 not contained in the FMECA and functionality analysis. There are components with materials different than those assumed in the FMECA; however, evaluations have been completed to verify that these differences do not affect the current aging management strategy. SGS Unit 1 meets the requirements for application of MRP-227-A as a strategy for managing age-related material degradation in reactor internals components. SGS Unit 1 will implement and apply the approved version of MRP-227 (MRP-227-A) as a strategy for managing age-related material degradation in reactor internals components [6].

6.2.2 SE Applicant/Licensee Action Item 2: PWR Vessel Internal Components within the Scope of License Renewal

As discussed in Section 3.2.5.2 of this SE, consistent with the requirements addressed in 10 CFR 54.4, each applicant/licensee is responsible for identifying which RVI components are within the scope of LR for its facility. Applicants/licensees shall review the information in Tables 4-1 and 4-2 in MRP-189, Revision 1, and Tables 4-4 and 4-5 in MRP-191 and identify whether these tables contain all of the RVI components that are within the scope of LR for their facilities in accordance with 10 CFR 54.4. If the tables do not identify all the RVI components that are within the scope of LR for its facility, the applicant or licensee shall identify the missing component(s) and propose any necessary modifications to the program defined in MRP-227, as modified by this SE, when submitting its plant-specific AMP. The AMP shall provide assurance that the effects of aging on the missing component(s) will be managed for the period of extended operation. This issue is Applicant/Licensee Action Item 2 [6].

SGS Unit 1 Compliance

This A/LAI requires comparison of the RVI components that are within the scope of license renewal for SGS Unit 1 to those components contained in MRP-191, Table 4-4. A detailed tabulation of the SGS Unit 1 RVI components was completed and compared favorably to the typical Westinghouse PWR internals components in MRP-191. Review of the information in Tables 4-1 and 4-2 in MRP-189, Revision 1 was not necessary as SGS Unit 1 is a Westinghouse design reactor.

Several components have different materials than those specified in the MRP-191 assessment but these have no effect on the recommended MRP aging management strategy; therefore, no modifications to the program details in MRP-227-A need to be proposed. An overview of those materials differences is provided here.

The guide plates/cards are considered in the MRP-191 assessment and identified as being 304 SS; however, for SGS Unit 1, the guide plates/cards have the potential to be cast CF8 material. An expert panel review was conducted, aligned with the process specified in MRP-191, Section 6 [14], and it was concluded with 100 percent consensus that there is no change to the current MRP conclusion that the guide plates/cards are a Primary inspection item [45]. Therefore, no modifications to the program detailed in MRP-227-A need to be proposed due to the assessment and screening of the RVI as developed in MRP-191 and MRP-232.

The brackets, clamps, terminal blocks and conduit straps (conduit support, conduit support gusset, gusset clamp, and thermocouple stop) are considered in the MRP-191 assessment and identified as being 304 SS; however, for SGS Unit 1, conduit support, conduit support gusset, gusset clamp, and thermocouple stop have the potential to be cast CF8 material. An expert panel review was conducted, aligned with the process specified in MRP-191, Section 6 [14], and it was concluded with 100 percent consensus that there is no change to the current MRP conclusion that the brackets, clamps, terminal blocks and conduit straps are not inspection components [45]. Therefore, no modifications to the program detailed in MRP-227-A need to be proposed due to the assessment and screening of the RVI as developed in MRP-191 and MRP-232.

This supports the requirement that the AMP shall provide assurance that the effects of aging on the SGS Unit 1 RVI components within the scope of license renewal but not included in the generic Westinghouse-designed RVI components from Table 4-4 of MRP-191 will be managed for the period of extended operation.

The generic scoping and screening of the RVI as summarized in MRP-191 and MRP-232 to support the inspection sampling approach for aging management of reactor internals specified in MRP-227-A is applicable to SGS Unit 1 with no modifications.

Conclusion

All components required to be included in the SGS Unit 1 program are consistent with those contained in MRP-191. Several components have materials different than those specified in MRP-191; however, evaluations have been completed to show that these differences have no effect on the MRP aging management strategy. SGS Unit 1 meets the requirement for application of MRP-227-A as a strategy for managing age-related degradation in reactor internals components.

6.2.3 SE Applicant/Licensee Action Item 3: Evaluation of the Adequacy of Plant-Specific Existing Programs

As addressed in Section 3.2.5.3 in this SE, applicants/licensees of CE and Westinghouse are required to perform plant-specific analysis either to justify the acceptability of an applicant's/licensee's existing programs, or to identify changes to the programs that should be implemented to manage the aging of these components for the period of extended operation. The results of this plant-specific analyses and a description of the plant-specific programs being relied on to manage aging of these components shall be submitted as part of the applicant's/licensee's AMP application. The CE and Westinghouse components identified for this type of plant-specific evaluation include: CE thermal shield positioning pins and CE in-core instrumentation thimble tubes (Section 4.3.2 in MRP-227), and Westinghouse guide tube support pins (split pins) (Section 4.3.3 in MRP-227). This is Applicant/Licensee Action Item 3 [6].

SGS Unit 1 Compliance

SGS Unit 1 is compliant with the requirements in Table 4-9 of MRP-227-A, as shown in Appendix C, Table C-3. This is detailed in the plant-specific SGS program documents for ASME Section XI [10] and the plant-specific flux thimble program [39].

In response to the industry concern, the control rod guide tube support pins fabricated from *INCONEL*® Alloy X-750 were replaced at SGS Unit 1 in the spring of 2007; the replacement support pins utilized improved materials (strain-hardened 316 stainless steel) that support the proactive management of aging in reactor internals components. Detailed descriptions of the replacement are retained in the plant records [38].

Conclusion

SGS Unit 1 complies with Applicant/Licensee Action Item 3 of the NRC SE on MRP-227, Revision 0, and therefore meets the requirement for application of MRP-227-A as a strategy for managing age-related material degradation in reactor internals components.

6.2.4 SE Applicant/Licensee Action Item 4: B&W Core Support Structure Upper Flange Stress Relief

As discussed in Section 3.2.5.4 of this SE, the B&W applicants/licensees shall confirm that the core support structure upper flange weld was stress relieved during the original fabrication of the Reactor Pressure Vessel in order to confirm the applicability of MRP-227, as approved by the NRC, to their facility. If the upper flange weld has not been stress relieved, then this component shall be inspected as a "Primary" inspection category component. If necessary, the examination methods and frequency for non-stress relieved B&W core support structure upper flange welds shall be consistent with the recommendations in MRP-227, as approved by the NRC, for the Westinghouse and CE upper core support barrel welds. The examination coverage for this B&W flange weld shall conform to the staff's imposed criteria as described in Sections 3.3.1 and 4.3.1 of this SE. The applicant's/licensee's resolution of this plant-specific action item shall be submitted to the NRC for review and approval. This is Applicant/Licensee Action Item 4 [6].

SGS Unit 1 Compliance

This Applicant/Licensee Action Item is not applicable to SGS Unit 1 since it is a Westinghouse design reactor, and this item only applies to B&W plants.

Conclusion

Applicant/Licensee Action Item 4 of the NRC SE on MRP-227, Revision 0 is not applicable to SGS Unit 1.

6.2.5 SE Applicant/Licensee Action Item 5: Application of Physical Measurements as part of I&E Guidelines for B&W, CE, and Westinghouse RVI Components

As addressed in Section 3.3.5 in this SE, applicants/licensees shall identify plant-specific acceptance criteria to be applied when performing the physical measurements required by the NRC-approved version of MRP-227 for loss of compressibility for Westinghouse hold down springs, and for distortion in the gap between the top and bottom core shroud segments in CE units with core barrel shrouds assembled in two vertical sections. The applicant/licensee shall include its proposed acceptance criteria and an explanation of how the proposed acceptance criteria are consistent with the plants' licensing basis and the need to maintain the functionality of the component being inspected under all licensing basis conditions of operation during the period of extended operation as part of their submittal to apply the approved version of MRP-227. This is Applicant/Licensee Action Item 5 [6].

SGS Unit 1 Compliance

SGS Unit 1 utilizes a Type 304 SS hold down spring; therefore, PSEG Nuclear is planning to perform inspections/physical measurements on the SGS Unit 1 hold down spring according to MRP-227-A. The proposed acceptance criteria for the physical measurements have been completed and are consistent with the licensing basis for SGS Unit 1 [46].

Acceptance criteria for distortion in the gap between the top and bottom core shroud segments are not applicable to SGS Unit 1 since it is a Westinghouse design reactor, and this element only applies to CE plants with core barrel shrouds assembled in two vertical sections.

Conclusion

SGS Unit 1 complies with Applicant/Licensee Action Item 5 of the NRC SE on MRP-227, Revision 0, and therefore meets the requirement for application of MRP-227-A as a strategy for managing age-related material degradation in reactor internals components.

6.2.6 SE Applicant/Licensee Action Item 6: Evaluation of Inaccessible B&W Components

As addressed in Section 3.3.6 in this SE, MRP-227 does not propose to inspect the following inaccessible components: the B&W core barrel cylinders (including vertical and circumferential seam welds), B&W former plates, B&W external baffle-to-baffle bolts and their locking devices, B&W core barrel-to-former bolts and their locking devices, and B&W core barrel assembly internal baffle-to-baffle bolts. The MRP also identified that although the B&W core barrel assembly internal baffle-to-baffle bolts are accessible, the bolts are non-inspectable using currently available examination techniques.

Applicants/licensees shall justify the acceptability of these components for continued operation through the period of extended operation by performing an evaluation, or by proposing a scheduled replacement of the components. As part of their application to implement the approved version of MRP-227, applicants/licensees shall provide their justification for the continued operability of each of the inaccessible components and, if necessary, provide their plan for the replacement of the components for NRC review and approval. This is Applicant/Licensee Action Item 6 [6].

SGS Unit 1 Compliance

This Applicant/Licensee Action Item is not applicable to SGS Unit 1 since it is a Westinghouse design reactor, and this item only applies to B&W plants.

Conclusion

Applicant/Licensee Action Item 6 of the NRC SE on MRP-227, Revision 0 is not applicable to SGS Unit 1.

6.2.7 SE Applicant/Licensee Action Item 7: Plant-Specific Evaluation of CASS Materials

As discussed in Section 3.3.7 of this SE, the applicants/licensees of B&W, CE, and Westinghouse reactors are required to develop plant-specific analyses to be applied for their facilities to demonstrate that B&W IMI guide tube assembly spiders and CRGT spacer castings, CE lower support columns, and Westinghouse lower support column bodies will maintain their functionality during the period of extended operation or for additional RVI components that may be fabricated from CASS, martensitic stainless steel or precipitation hardened stainless steel materials. These analyses shall also consider the possible loss of fracture toughness in these components due to thermal and irradiation embrittlement, and may also need to consider limitations on accessibility for inspection and the resolution/sensitivity of the inspection techniques. The requirement may not apply to components that were previously evaluated as not requiring aging management during development of MRP-227. That is, the requirement would apply to components fabricated from susceptible materials for which an individual licensee has determined aging management is required, for example during their review performed in accordance with Applicant/Licensee Action Item 2. The plant-specific analysis shall be consistent with the plant's licensing basis and the need to maintain the functionality of the components being evaluated under all licensing basis conditions of operation. The applicant/licensee shall include the plant-specific analysis as part of their submittal to apply the approved version of MRP-227. This is Applicant/Licensee Action Item 7 [6].

SGS Unit 1 Compliance

A/LAI 7, from the NRC's final SE on MRP-227, revision 0, states that, for assessment of CASS materials, the applicant/licensee for license renewal may apply the criteria detailed in [43] as the basis for determining whether the CASS materials are susceptible to the thermal aging mechanism. If the application of the applicable screening criteria for the component's material demonstrates that the components are not susceptible to either thermal embrittlement (TE) or irradiation embrittlement (IE), or to the synergistic effects of TE and IE combined, then no other evaluation would be necessary. The SGS Unit 1 CASS RVI components and the assessment of their susceptibility to TE are summarized in Table 6-2. The B&W and CE components fabricated of CASS material, as discussed in A/LAI 7, are not applicable to SGS Unit 1 since it is a Westinghouse design reactor.

Using the chemistry data from retrieved certified material test reports (CMTRs) as input into Hull's formula per the guidance of [47], the ferrite content was determined. Based on the criteria of [43], 23 of the SGS Unit 1 CASS (Grade CF8) lower support column caps are not susceptible to TE. CMTRs were not located for 73 of the SGS Unit 1 CASS lower support column caps; therefore, it is conservatively assumed that they are potentially susceptible to TE.

The lower internals assembly column cap is a CASS piece welded onto the top of the core support column shaft. These two pieces together constitute the lower internals assembly – column body. The development of the MRP-227-A aging management strategy considered the lower support column as one complete unit denoted as the "lower support column assemblies – lower support column bodies." Under the lower support column bodies in MRP-191, both Type 304 SS and CF8 CASS material were considered. Since the lower internals assembly column cap is part of the lower support column body, it was addressed by the generic industry FMECA and functionality analysis; therefore, it is subject to the same inspection

requirements as the lower support column bodies (cast). The inspection of the lower support column body includes the column cap.

As shown in Table 6-2, the remaining SGS Unit 1 components have Grade CF8 specified as either the primary material or as an alternate material on their engineering drawings. Where alternate CF8 is permitted, in the evaluation supporting A/LAI 7, it is conservative to assume that the material used for these components is Grade CF8. Confirmation of material composition under TE susceptibility thresholds was not demonstrated for these remaining components; therefore, it is conservatively assumed that they are potentially susceptible to TE.

In the development of MRP-191, the control rod guide tube assemblies - guide plates/cards and the upper instrumentation conduit and supports - brackets, clamps, terminal blocks and conduit straps were screened as wrought material (304 SS). The material difference for the corresponding SGS Unit 1 CASS guide plates/cards and SGS Unit 1 CASS conduit support, conduit support gusset, gusset clamp, and thermocouple stop is addressed in Sections 6.2.1 and 6.2.2.

Irradiation may also cause a material to undergo embrittlement. In MRP-191, the guide tube lower flange, mixing device, upper support column base, and lower support column each screened-in as susceptible to IE. The guide tube intermediate flange, guide plate, upper instrumentation supports, and core support casting screened below the MRP-191 fluence screening level; thus, they are not susceptible to IE [45].

No martensitic SS or martensitic PH-SS components were identified for the SGS Unit 1 RVI.

Conclusion

It is concluded that continued application of the MRP-227-A strategy will meet the requirement for managing age-related degradation of the SGS Unit 1 CASS RVI components.

Table 6-2 Summary of SGS Unit 1 CASS Components and their Susceptibility to TE

CASS Component	SGS RVI AMR Component Name [1]	Molybdenum Content	Casting	Ferrite Content	Susceptibility to TE (Based on the NRC Criteria [43])
Upper Internals Assembly					
17 x 17 Guide Tube Intermediate Flange	Not Applicable(3)	0.5 Maximum	Static	Possible > 20%(1)	Potentially Susceptible
17 x 17 Guide Tube Lower Flange	RCCA Guide Tube Assemblies (lower flanges)	0.5 Maximum	Static	Possible > 20%(1)	Potentially Susceptible
17 x 17 Guide Tube Lower Guide Plates/Cards	RCCA guide tube assemblies (tubes, housing plates, and guide plates)	0.5 Maximum	Static	Possible > 20%(1)	Potentially Susceptible
Top Mounted Mixing Device, Mixing Device	Upper Internals Assembly (static flow mixers)	0.5 Maximum	Static	Possible > 20%(1)	Potentially Susceptible
Upper Instrumentation Conduit and Supports - Gussets, Clamps, Supports, and Thermocouple Stop	Not Applicable(3)	0.5 Maximum	Static	Possible > 20%(1)	Potentially Susceptible
Upper Support Column - Orifice Base	Upper Internals Assembly (upper support column bases)	0.5 Maximum	Static	Possible > 20%(1)	Potentially Susceptible
Upper Support Column - Mixing Bases	Upper Internals Assembly (static flow mixers)	0.5 Maximum	Static	Possible > 20%(1)	Potentially Susceptible
Lower Internals Assembly					
Lower Support Column - Cap	Lower Internal Assembly (core support, incl'g core support lugs, columns and sleeves)	0.5 Maximum	Static	≤ 20%(2) Possible > 20%(1)	23 of 96 Are Not Susceptible 73 Potentially Susceptible
Core Support Casting	Lower Internal Assembly (core support dome)	0.5 Maximum	Static	Possible > 20%(1)	Potentially Susceptible

Notes:
1. Where component-specific CMTR is not available, the ferrite content is calculated. Allowable variants of the material specification chemistry requirements may result in ferrite content estimations higher than 20%; thus, the ferrite content is identified as potentially exceeding 20%.
2. Conclusion is based on CMTR chemistry data. The CMTRs do not list the element percentage for nitrogen; thus, per the guidance of [47], nitrogen is assumed to be 0.04 percent. The CMTRs do not list an elemental percentage for molybdenum. Therefore, the current Grade CF8 chemistry requirements were reviewed which specify a maximum of 0.5 percent molybdenum. This maximum value is input into Hull's formula
3. Component was not evaluated within the SGS Unit 1 AMR [1]; however, because it is manufactured with CASS material, it has been evaluated for aging within this AMP.

6.2.8 SE Applicant/Licensee Action Item 8: Submittal of Information for Staff Review and Approval

As addressed in Section 3.5.1 in this SE, applicants/licensees shall make a submittal for NRC review and approval to credit their implementation of MRP-227, as amended by this SE, as an AMP for the RVI components at their facility. This submittal shall include the information identified in Section 3.5.1 of this SE. This is Applicant/Licensee Action Item 8 [6].

SGS Unit 1 Compliance

SGS Unit 1, per the RIS [4], is considered a Category B plant that is expected to submit their RVI AMP/inspection plan based on the guidance of MRP-227-A, consistent with their commitments. Per the LRA [1], SGS Unit 1 has a commitment to submit their RVI inspection plan for approval by the NRC no later than August 13, 2014. The SGS Unit 1 RVI AMP will also be submitted to meet the guidelines provided in the RIS [4].

Conclusion

SGS Unit 1 complies with Applicant/Licensee Action Item 8 of the NRC SE on MRP-227, Revision 0, and therefore meets the requirement for application of MRP-227-A as a strategy for managing age-related material degradation in reactor internals components.

7 PROGRAM ENHANCEMENT AND IMPLEMENTATION SCHEDULE

The requirements of MRP-227-A are based on an 18-month refueling cycle and consider both EFPY and cumulative operation. The information contained in Table 7-1 is based on this information and includes a description of the latest scope of inspections pertaining to the reactor internals AMP. Should a change occur in plant operational practices or should operating experience result in changes to the projections, appropriate updates will be performed on affected plant documentation in accordance with approved procedures.

Table 7-1 SGS Unit 1 Aging Management Program Enhancement and Inspection Implementation Summary

Refueling Outage	Project Month/Year	Estimated EFPY	AMP-Related Scope	Inspection Method and Criteria	Comments
21	Fall 2011	22.8	Not applicable	Not applicable	Not applicable
22	Spring 2013	24.3	Not applicable	Not applicable	Not applicable
23	Fall 2014	25.7	Not applicable	Not applicable	Not applicable
24	Spring 2016	27.2	Not applicable	Not applicable	**Period of Extended Operation begins on August 13, 2016**
25	Fall 2017	28.6	Not applicable	Not applicable	Not applicable
26	Spring 2019	30.1	ASME Code Section XI 10-Year ISI. Initial MRP-227-A augmented inspection for control rod guide tube lower flange welds, upper and lower core barrel flange welds, upper and lower core barrel cylinder girth welds, and thermal shield flexures completed during or before this outage.	ASME Code Section XI MRP-227-A inspections in accordance with MRP-228 specifications	SGS Unit 1 will begin the period of extended operation during Cycle 25 (post 1RF24). PSEG Nuclear has the option to perform these initial augmented inspections until 1RF26. The inspection window for these components is within two refueling cycles from the beginning of extended operation.

Table 7-1 (cont.) SGS Unit 1 Aging Management Program Enhancement and Inspection Implementation Summary

Refueling Outage	Project Month/Year	Estimated EFPY	AMP-Related Scope	Inspection Method and Criteria	Comments
27	Fall 2020	31.5	Initial MRP-227-A augmented inspection for control rod guide tube guide plates (cards) and internals hold down spring completed during or before this outage.	MRP-227-A inspections in accordance with MRP-228 specifications	The inspection window for 17x17 standard guide tubes in Westinghouse four-loop plants is 24 to 28 EFPY. As SGS Unit 1 was a participating plant for this analysis, an additional four EFPY can be applied to the initial inspection measurement schedule. Therefore, the initial inspection must be performed before SGS Unit 1 reaches 32 EFPY. Subsequent inspections are based on initial inspection results. See WCAP-17451-P [44] for additional information regarding the inspection schedule and requirements. The inspection window for the hold down spring is within three refueling cycles from the beginning of extended operation. If spring height is not sufficient, spring height measurements must be taken during the next two outages.
28	Spring 2022	33.0	Not applicable	Not applicable	Not applicable
29	Fall 2023	34.4	Initial MRP-227-A augmented inspection for baffle-former bolts completed during or before this outage.	MRP-227-A inspections in accordance with MRP-228 specifications	The inspection window for baffle-former bolts is between 25 and 35 EFPY. PSEG Nuclear has the option to perform this

Table 7-1 (cont.) SGS Unit 1 Aging Management Program Enhancement and Inspection Implementation Summary

Refueling Outage	Project Month/Year	Estimated EFPY	AMP-Related Scope	Inspection Method and Criteria	Comments
					inspection until RO-29.
30	Spring 2025	35.9	Not applicable	Not applicable	Not applicable
31	Fall 2026	37.3	Not applicable	Not applicable	Not applicable
32	Spring 2028	38.8	Initial MRP-227-A augmented inspections for baffle-former assembly and baffle-edge bolts completed during or before this outage.	MRP-227-A inspections in accordance with MRP-228 specifications	The inspection window for these components is between 20 and 40 EFPY. PSEG Nuclear has the option to perform these inspections until RO-32.
33	Fall 2029	40.2	ASME Code Section XI 10-Year ISI. Subsequent MRP-227-A augmented inspections for control rod guide tube lower flange welds, upper and lower core barrel flange welds, upper and lower core barrel cylinder girth welds, and thermal shield flexures completed on a ten-year interval.	MRP-227-A inspections in accordance with MRP-228 specifications	The inspection window for these components is 10 years after the initial inspection.
34	Spring 2031	41.7	Not applicable	Not applicable	Not applicable
35	Fall 2032	43.1	Subsequent MRP-227-A augmented inspection for baffle-former bolts completed on a ten-year interval.	MRP-227-A inspections in accordance with MRP-228 specifications	The inspection window for these components is 10 years after the initial inspection.
36	Spring 2034	44.6	Not applicable	Not applicable	Not applicable
37	Fall 2035	46.0	Not applicable	Not applicable	Not applicable
38	August 13, 2036	47	Not applicable	Not applicable	Renewed Operating License expires August 13, 2036

8 IMPLEMENTING DOCUMENTS

The SGS Unit 1 PWR Vessel Internals AMP is implemented through PSEG procedure ER-AP-333, "Pressurized Water Reactor Internals Aging Management Program", Rev. 0 [2]. It also credits existing aging management programs. The SGS Unit 1 RVI AMP references the Water Chemistry Program and the ASME Code Section XI ISI, Subsections IWB, IWC, and IWD Program. MRP-227-A augmented examinations (Appendix C), recommended as a result of industry programs, will be included in the existing ASME Section XI Program.

PSEG documents associated with the existing SGS Programs and considered to be implementing documents of the SGS Unit 1 PWR Vessel Internals Program are:

- CY-AP-120-1000, Primary Strategic Water Chemistry Plan for Recirculating Steam Generator Plants [19]
- CY-AP-120-100, Reactor Coolant System Chemistry [20]
- ER-AA-330, Conduct of Inservice Inspection Activities [10]
- SA-PBD-AMP-X.M1, Metal Fatigue of Reactor Coolant Pressure Boundary [40]

The SGS Unit 1 RVI AMP relies on the Water Chemistry Program for maintaining high water purity to reduce susceptibility to cracking due to SCC. The Water Chemistry Program was evaluated and found to be consistent with GALL Section XI.M2, Water Chemistry [18]. Additional procedures may be updated or created as OE for augmented examinations is accumulated.

Based on this information, the updated AMP for SGS Unit 1 RVI provides reasonable assurance that the aging effects will be managed such that the components within the scope of license renewal will continue to perform their intended functions consistent with the CLB for the period of extended operation.

9 REFERENCES

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8. SGS Units 1 and 2, Program Basis Document SA-PBD-AMP-XI.M16, Revision 3, "GALL Program XI.M16 - PWR Vessel Internals," January 14, 2011.
9. PSEG Report, Revision 27, "Salem Nuclear Generating Station Unit 1 and Unit 2 Updated Final Safety Analysis Report," November 25, 2013.
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17. NEI 03-08, Revision 2, "Guidelines for the Management of Materials Issues," Nuclear Energy Institute, Washington, DC, January 2010.

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19. SGS Document CY-AP-120-1000, Revision 8, "Primary Strategic Water Chemistry Plan for Recirculating Steam Generator Plants," June 10, 2013.

20. SGS Document CY-AP-120-100, Revision 16, "Reactor Coolant System Chemistry," May 9, 2013.

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41. SGS Transmittal of Design Information, S-TODI-2013-0003, Revision 1, "Westinghouse Request for Information in Support of PSEG for PA-MSC-0983," December 12, 2013.

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APPENDIX A
ILLUSTRATIONS

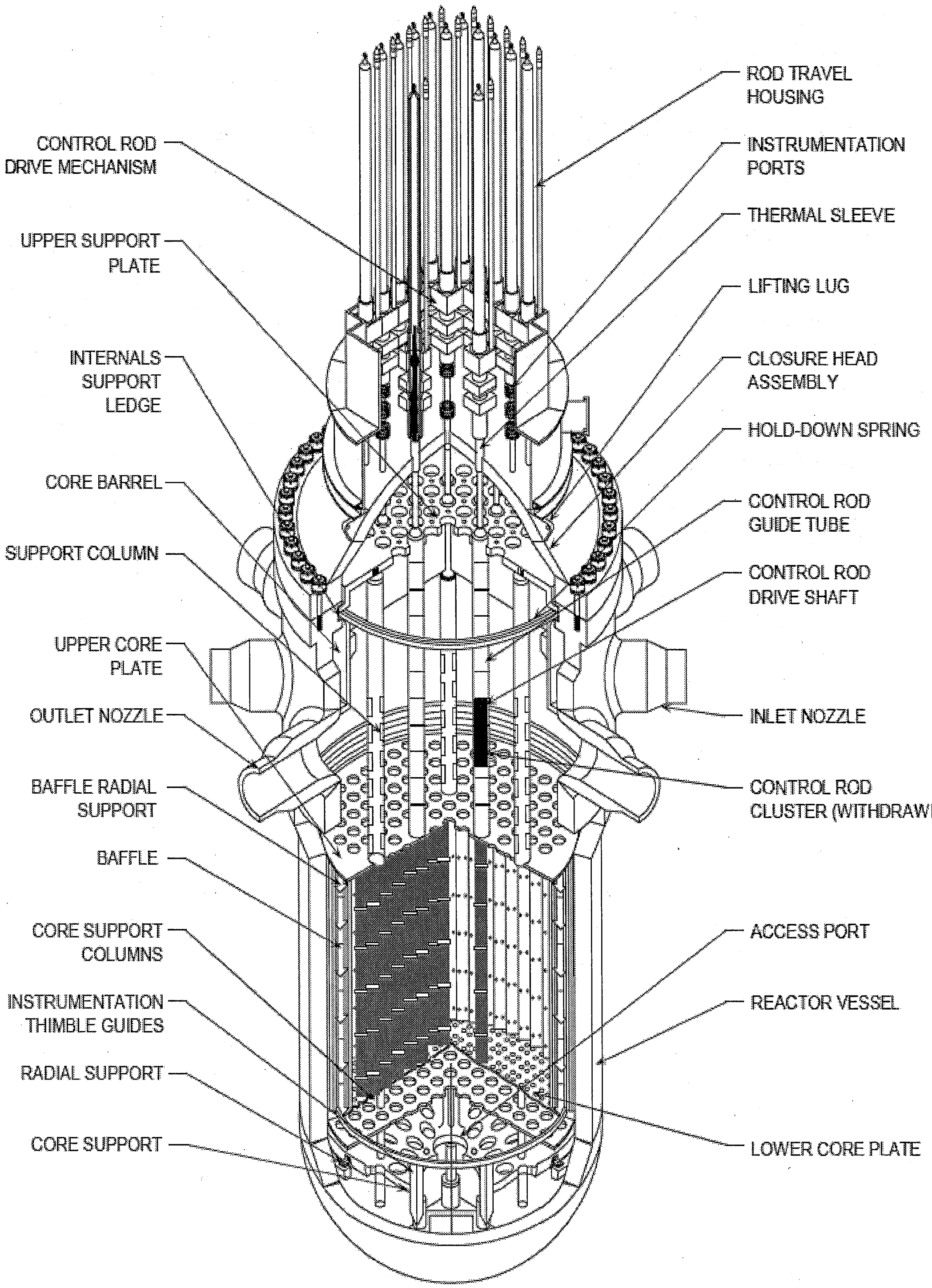

Figure A-1 Illustration of Typical Westinghouse Internals

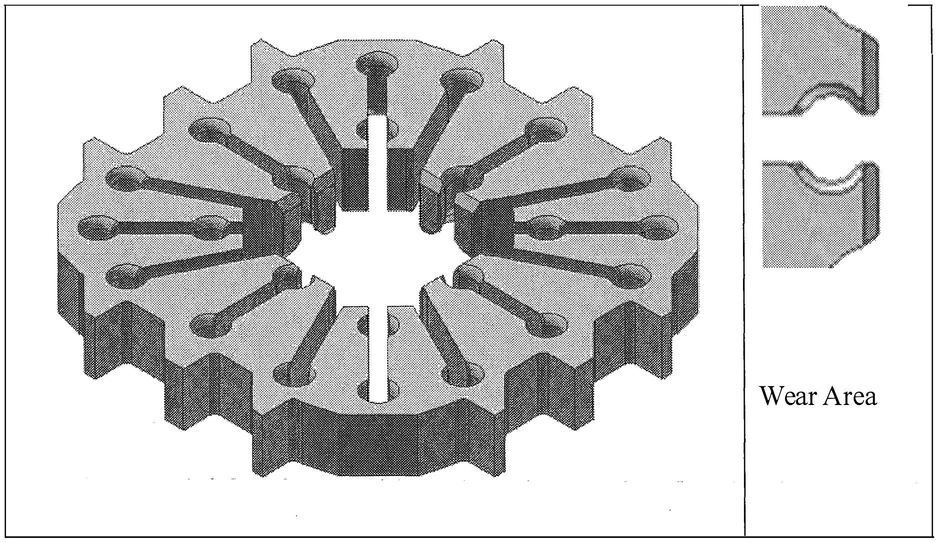

Figure A-2 Typical Westinghouse Control Rod Guide Card

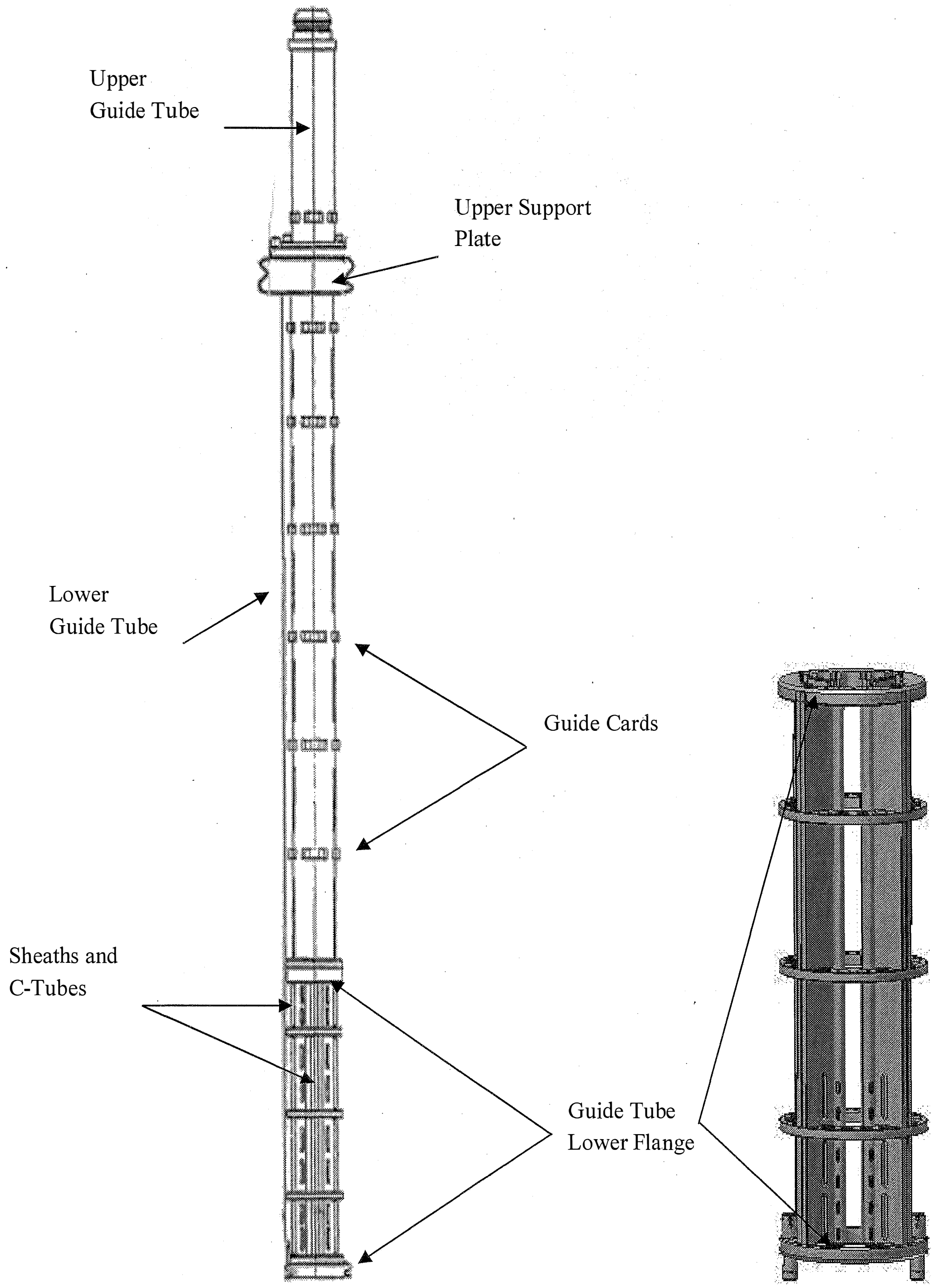

Figure A-3 Lower Section of Control Rod Guide Tube Assembly

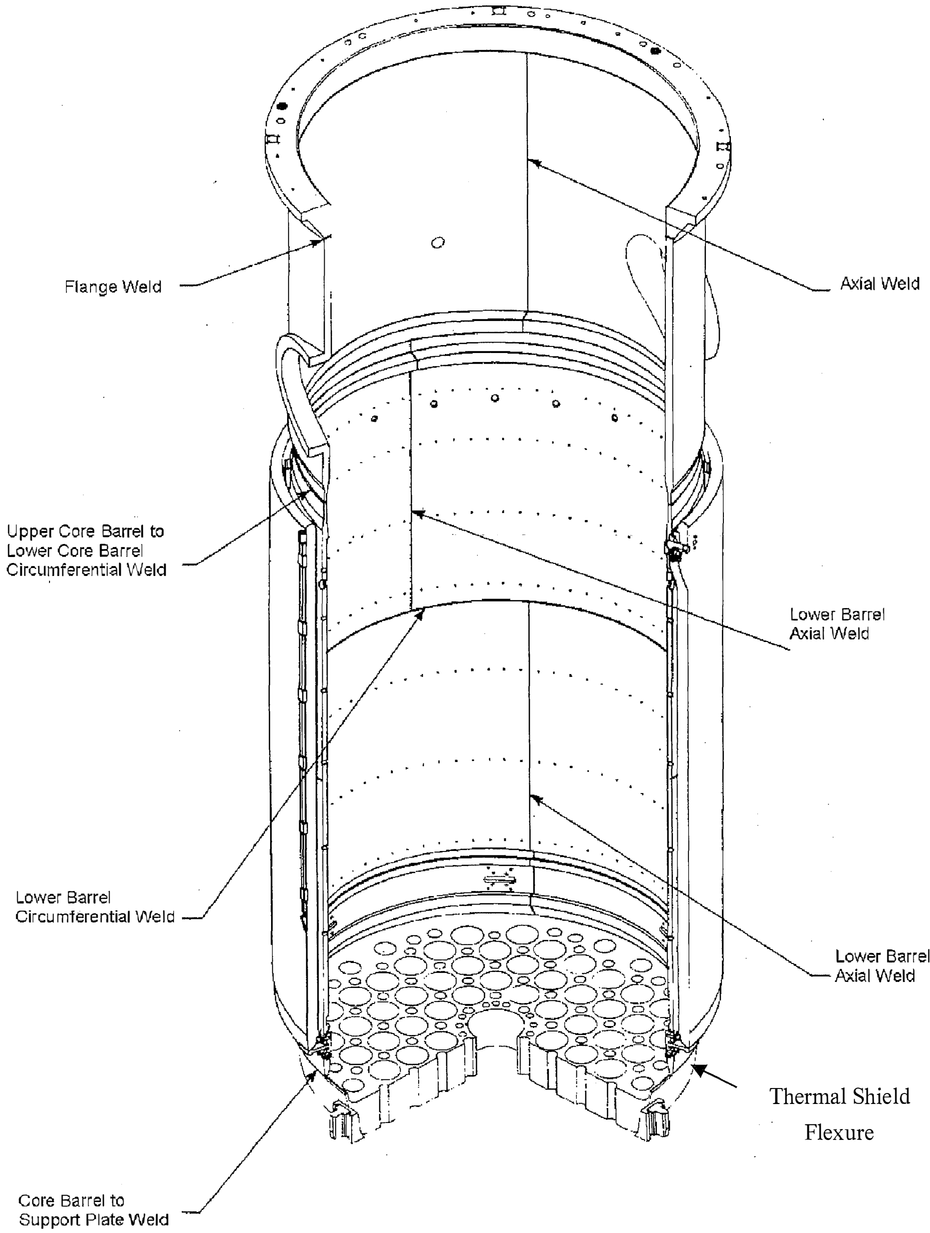

Figure A-4 Major Core Barrel Welds

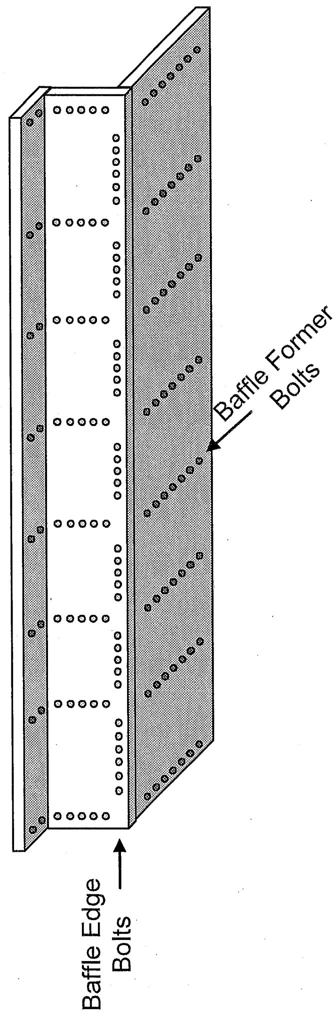

Figure A-5 Bolting Systems used in Typical Westinghouse Core Baffles
(Baffle-edge bolts are applicable for SGS Unit 1)

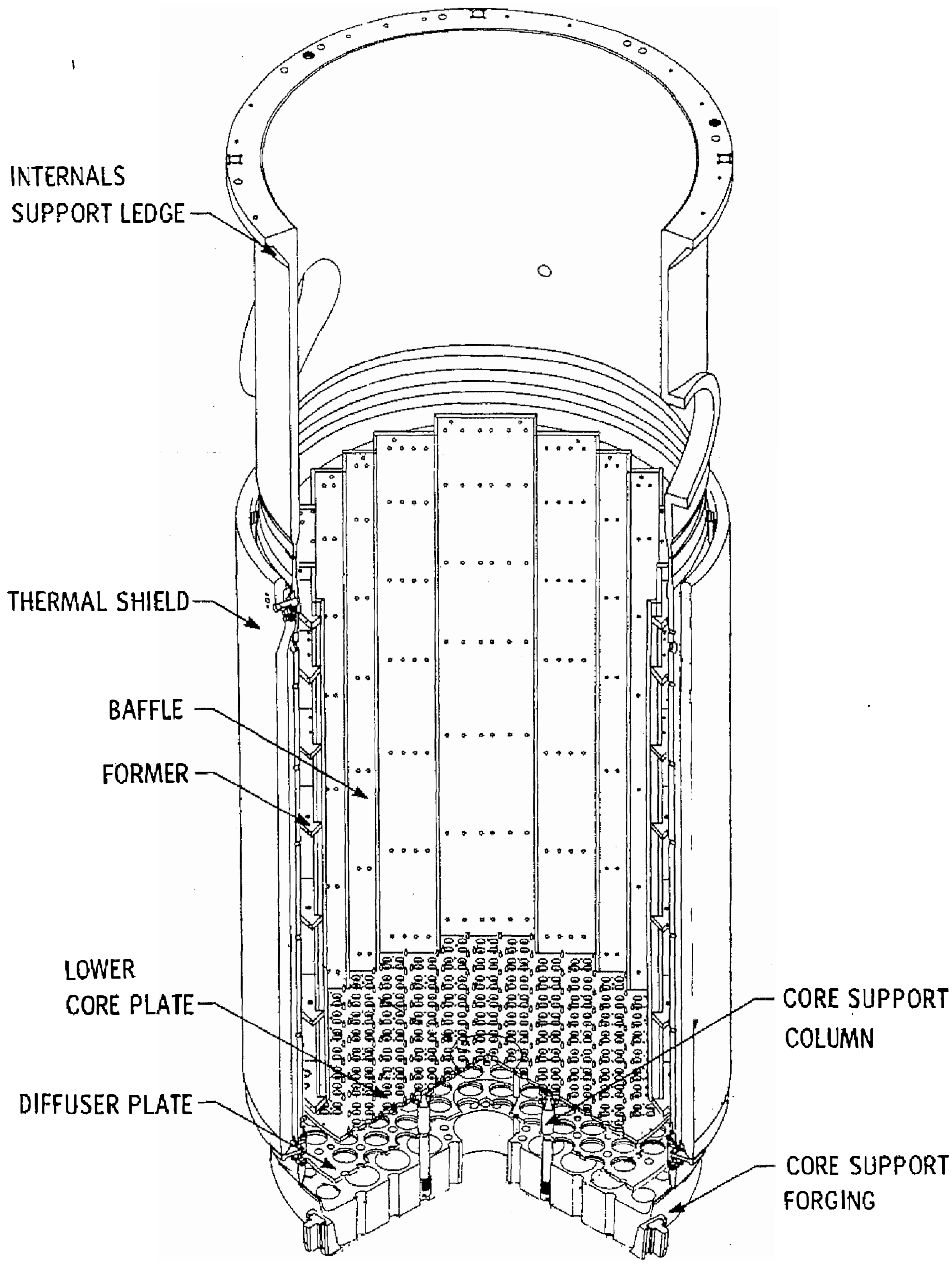

Figure A-6 Core Baffle/Barrel Structure

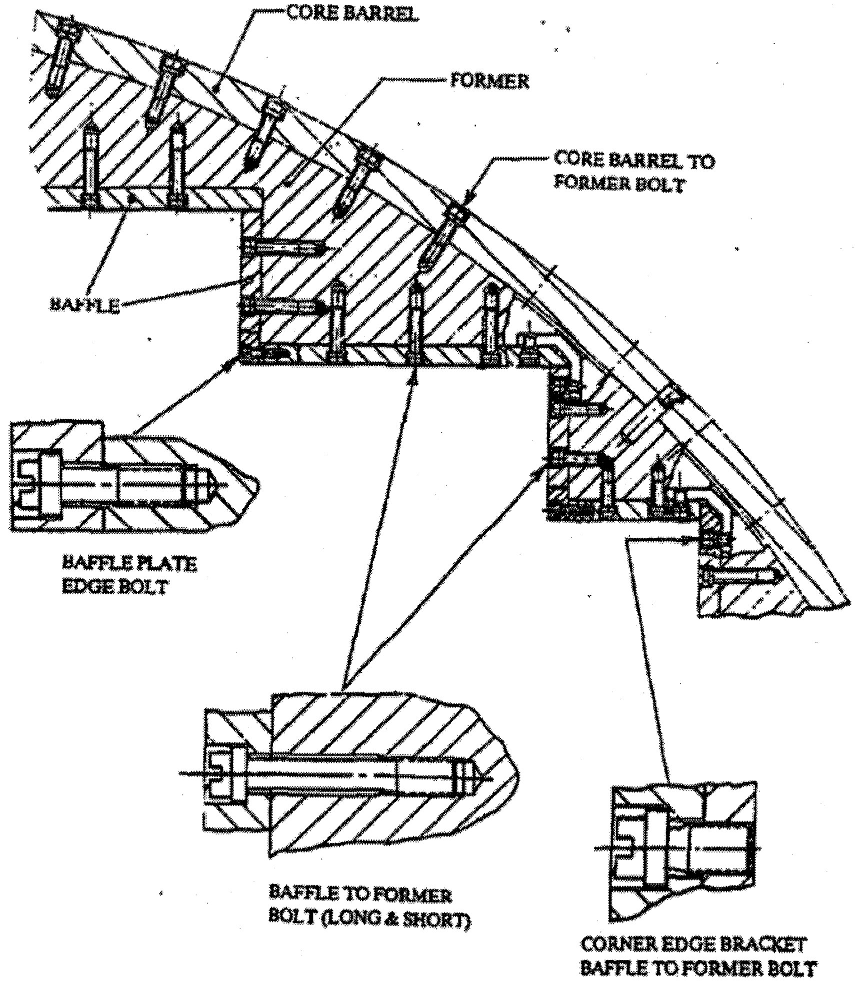

Figure A-7 Bolting in a Typical Westinghouse Baffle/Former Structure
(Baffle-edge bolts are applicable for SGS Unit 1)

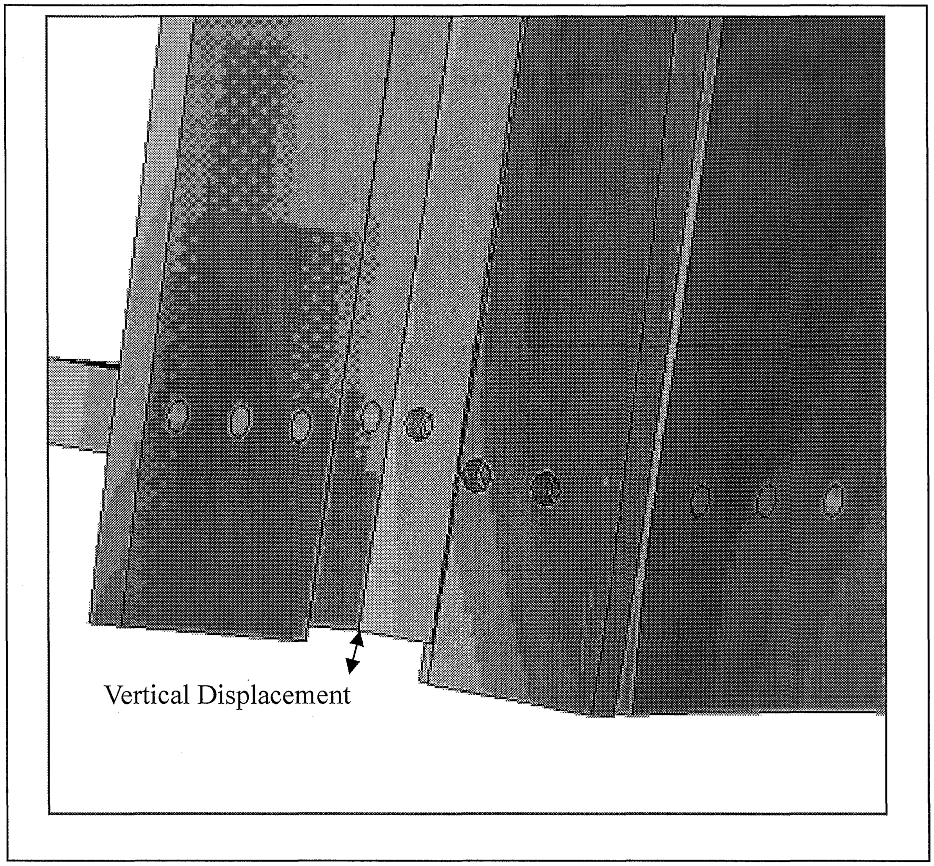

Figure A-8 Example of the Vertical Displacement between the Baffle Plates and Bracket at the Bottom of the Baffle-Former-Barrel Assembly which could Result from Void Swelling

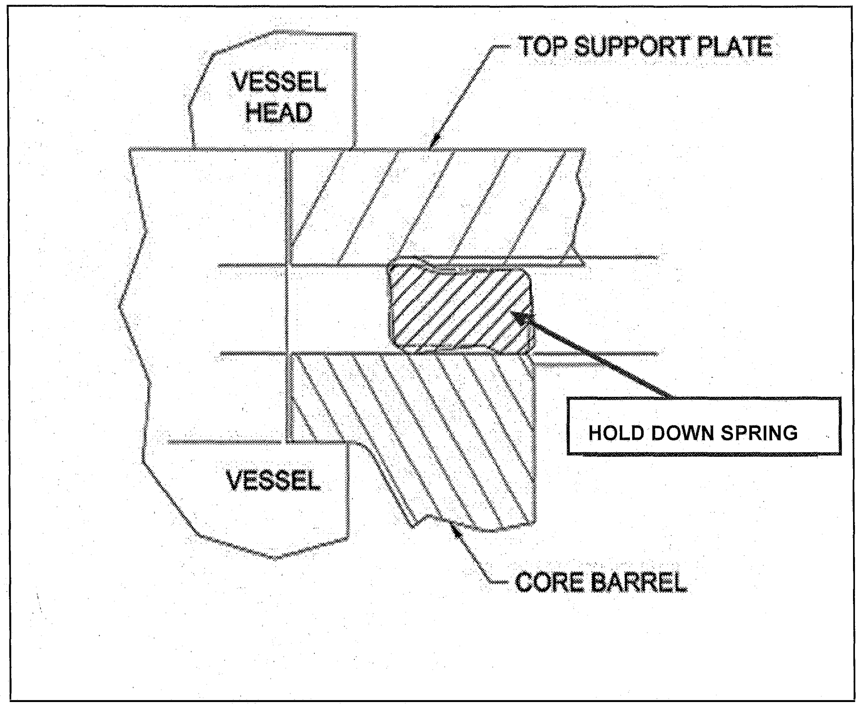

Figure A-9 Schematic Cross-Sections of the Westinghouse Hold Down Spring

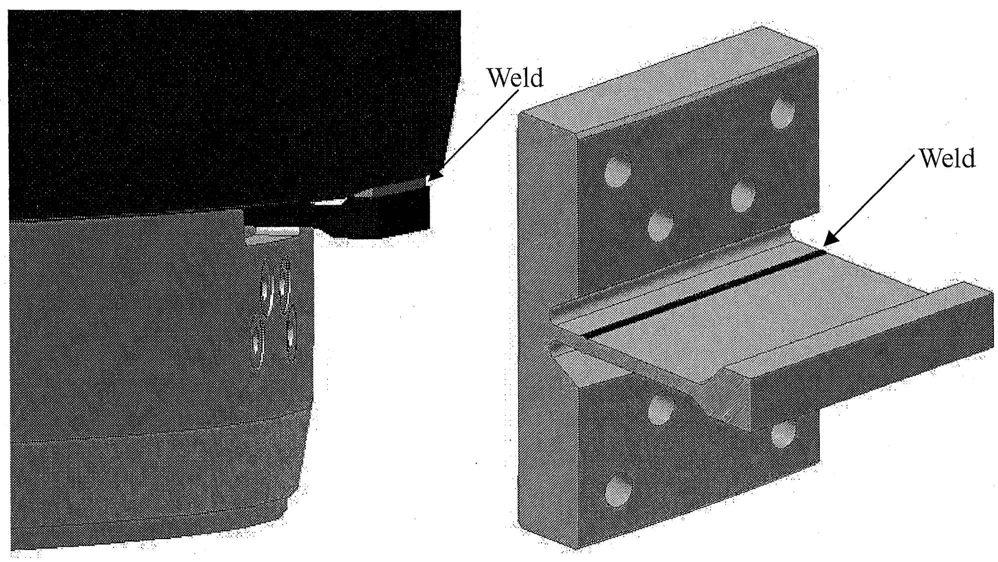

Figure A-10 Typical Thermal Shield Flexure

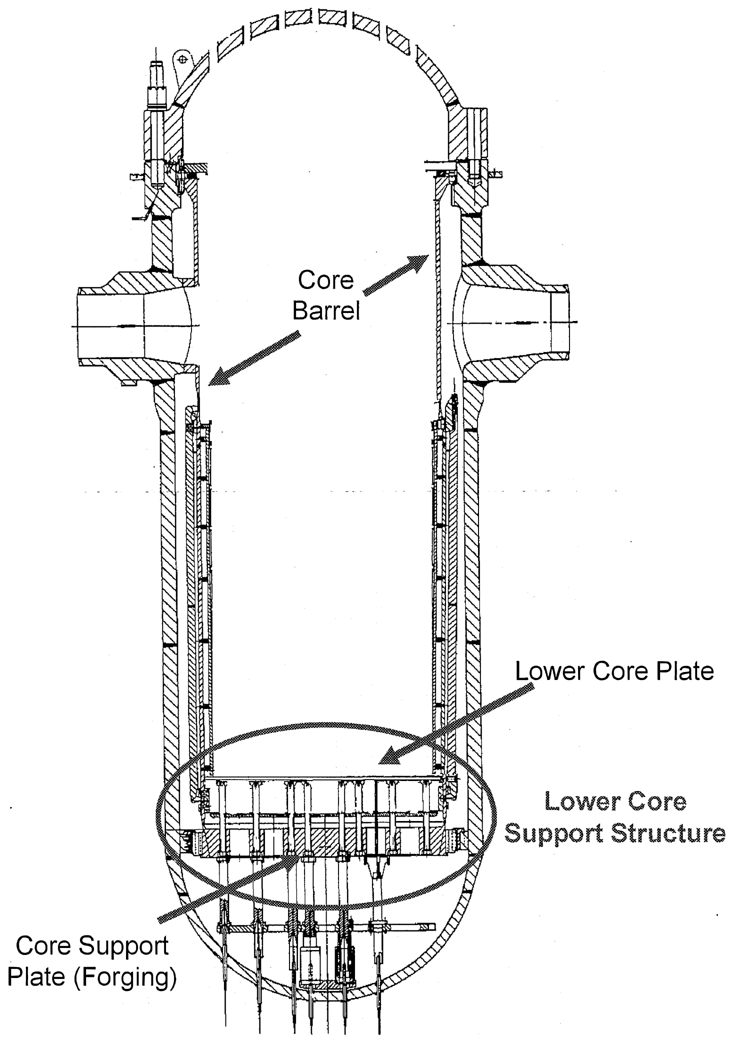

Figure A-11 Lower Core Support Structure

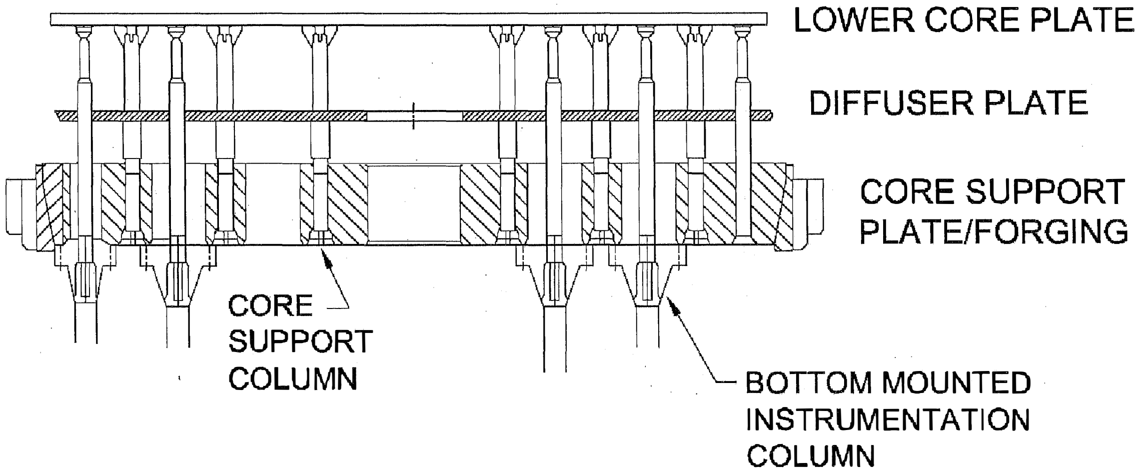

Figure A-12 Lower Core Support Structure – Cross-Section

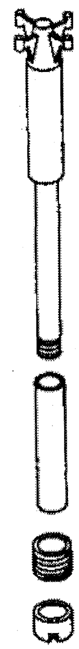

Figure A-13 Typical Core Support Column

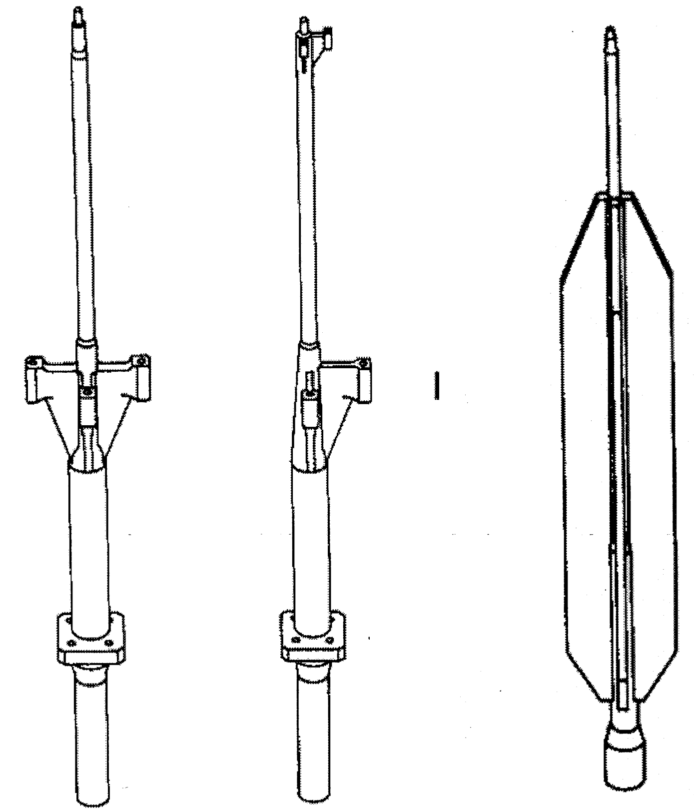

Figure A-14 Examples of BMI Column Designs

APPENDIX B
SALEM NUCLEAR GENERATING STATION
LICENSE RENEWAL AGING MANAGEMENT REVIEW
SUMMARY TABLES

The content in Table B-1 is extracted from Table 3.1.2-3 "Reactor Vessel Internals Summary of Aging Management Evaluation" of the SGS LRA [1]. Only those items applicable to RVI (according to the LRA) were imported into Table B-1 from the LRA. Per reference [41], it was identified that some components listed in this table were not consistent with the RVI design for SGS Unit 1. Notes have been added in the comments column to identify these components within Table B-1.

Table B-1 Summary of SGS LRA Table 3.1.2-3 – Reactor Vessel Internals Summary of Aging Management Evaluation

Component Type	Material	Aging Effect Requiring Management	Aging Management Programs	Comments
Control Rod Assembly	Not Applicable	Not Applicable	Not Applicable	Not Applicable
Core Barrel (lower)	Stainless Steel	Changes in Dimensions/Void Swelling	PWR Vessel Internals	Not Applicable
	Stainless Steel	Cracking/Stress Corrosion Cracking, Irradiation-Assisted Stress Corrosion Cracking	PWR Vessel Internals	Not Applicable
	Stainless Steel	Cracking/Stress Corrosion Cracking, Irradiation-Assisted Stress Corrosion Cracking	Water Chemistry	Not Applicable
	Stainless Steel	Loss of Fracture Toughness/Neutron Irradiation Embrittlement, Void Swelling	PWR Vessel Internals	Not Applicable
	Stainless Steel	Loss of Material/Pitting and Crevice Corrosion	Water Chemistry	Not Applicable

Table B-1 Summary of SGS LRA Table 3.1.2-3 – Reactor Vessel Internals Summary of Aging Management Evaluation

(cont.)

Component Type	Material	Aging Effect Requiring Management	Aging Management Programs	Comments
Core Barrel (upper)	Stainless Steel	Changes in Dimensions/Void Swelling	PWR Vessel Internals	Not Applicable
	Stainless Steel	Cracking/Stress Corrosion Cracking, Irradiation-Assisted Stress Corrosion Cracking	PWR Vessel Internals	Not Applicable
	Stainless Steel	Cracking/Stress Corrosion Cracking, Irradiation-Assisted Stress Corrosion Cracking	Water Chemistry	Not Applicable
	Stainless Steel	Loss of Fracture Toughness/Neutron Irradiation Embrittlement, Void Swelling	PWR Vessel Internals	Not Applicable
	Stainless Steel	Loss of Material/Pitting and Crevice Corrosion	Water Chemistry	Not Applicable
Core Barrel Assembly (alignment pins)	Stainless Steel	Changes in Dimensions/Void Swelling	PWR Vessel Internals	Not Applicable
	Stainless Steel	Cracking/Stress Corrosion Cracking, Irradiation-Assisted Stress Corrosion Cracking	PWR Vessel Internals	Not Applicable
	Stainless Steel	Cracking/Stress Corrosion Cracking, Irradiation-Assisted Stress Corrosion Cracking	Water Chemistry	Not Applicable
	Stainless Steel	Loss of Fracture Toughness/Neutron Irradiation Embrittlement, Void Swelling	PWR Vessel Internals	Not Applicable
	Stainless Steel	Loss of Material/Pitting and Crevice Corrosion	Water Chemistry	Not Applicable
	Stainless Steel	Loss of Preload/Stress Relaxation	PWR Vessel Internals	Not Applicable

Table B-1 Summary of SGS LRA Table 3.1.2-3 – Reactor Vessel Internals Summary of Aging Management Evaluation

(cont.)

Component Type	Material	Aging Effect Requiring Management	Aging Management Programs	Comments
Core Barrel Assembly (baffle bolt lock bars)	Stainless Steel	Changes in Dimensions/Void Swelling	PWR Vessel Internals	Not Applicable
	Stainless Steel	Cracking/Stress Corrosion Cracking, Irradiation-Assisted Stress Corrosion Cracking	PWR Vessel Internals	Not Applicable
	Stainless Steel	Cracking/Stress Corrosion Cracking, Irradiation-Assisted Stress Corrosion Cracking	Water Chemistry	Not Applicable
	Stainless Steel	Loss of Fracture Toughness/Neutron Irradiation Embrittlement, Void Swelling	PWR Vessel Internals	Not Applicable
	Stainless Steel	Loss of Material/Pitting and Crevice Corrosion	Water Chemistry	Not Applicable
	Stainless Steel	Loss of Preload/Stress Relaxation	PWR Vessel Internals	Not Applicable
Core Barrel Assembly (baffle former assembly - plates)	Stainless Steel	Changes in Dimensions/Void Swelling	PWR Vessel Internals	Not Applicable
	Stainless Steel	Cracking/Stress Corrosion Cracking, Irradiation-Assisted Stress Corrosion Cracking	PWR Vessel Internals	Not Applicable
	Stainless Steel	Cracking/Stress Corrosion Cracking, Irradiation-Assisted Stress Corrosion Cracking	Water Chemistry	Not Applicable
	Stainless Steel	Loss of Fracture Toughness/Neutron Irradiation Embrittlement, Void Swelling	PWR Vessel Internals	Not Applicable
	Stainless Steel	Loss of Material/Pitting and Crevice Corrosion	Water Chemistry	Not Applicable

Table B-1 (cont.) **Summary of SGS LRA Table 3.1.2-3 – Reactor Vessel Internals Summary of Aging Management Evaluation**

Component Type	Material	Aging Effect Requiring Management	Aging Management Programs	Comments
Core Barrel Assembly (core barrel to thermal shield bolts and dowels)	Stainless Steel Bolting	Changes in Dimensions/Void Swelling	PWR Vessel Internals	Not Applicable
	Stainless Steel Bolting	Cracking/Stress Corrosion Cracking, Irradiation-Assisted Stress Corrosion Cracking	PWR Vessel Internals	Not Applicable
	Stainless Steel Bolting	Cracking/Stress Corrosion Cracking, Irradiation-Assisted Stress Corrosion Cracking	Water Chemistry	Not Applicable
	Stainless Steel Bolting	Loss of Fracture Toughness/Neutron Irradiation Embrittlement, Void Swelling	PWR Vessel Internals	Not Applicable
	Stainless Steel Bolting	Loss of Material/Pitting and Crevice Corrosion	Water Chemistry	Not Applicable
	Stainless Steel Bolting	Loss of Preload/Stress Relaxation	PWR Vessel Internals	Not Applicable
Core Barrel Assembly (flange)	Stainless Steel	Changes in Dimensions/Void Swelling	PWR Vessel Internals	Not Applicable
	Stainless Steel	Cracking/Stress Corrosion Cracking, Irradiation-Assisted Stress Corrosion Cracking	PWR Vessel Internals	Not Applicable
	Stainless Steel	Cracking/Stress Corrosion Cracking, Irradiation-Assisted Stress Corrosion Cracking	Water Chemistry	Not Applicable
	Stainless Steel	Loss of Fracture Toughness/Neutron Irradiation Embrittlement, Void Swelling	PWR Vessel Internals	Not Applicable
	Stainless Steel	Loss of Material/Pitting and Crevice Corrosion	Water Chemistry	Not Applicable

Table B-1 Summary of SGS LRA Table 3.1.2-3 – Reactor Vessel Internals Summary of Aging Management Evaluation

(cont.)

Component Type	Material	Aging Effect Requiring Management	Aging Management Programs	Comments
Core Barrel Assembly (lock bar, baffle-former, barrel-former, and baffle-edge bolting)	Stainless Steel Bolting	Changes in Dimensions/Void Swelling	PWR Vessel Internals	Not Applicable
	Stainless Steel Bolting	Cracking/Stress Corrosion Cracking, Irradiation-Assisted Stress Corrosion Cracking	PWR Vessel Internals	Not Applicable
	Stainless Steel Bolting	Cracking/Stress Corrosion Cracking, Irradiation-Assisted Stress Corrosion Cracking	Water Chemistry	Not Applicable
	Stainless Steel Bolting	Loss of Fracture Toughness/Neutron Irradiation Embrittlement, Void Swelling	PWR Vessel Internals	Not Applicable
	Stainless Steel Bolting	Loss of Material/Pitting and Crevice Corrosion	Water Chemistry	Not Applicable
	Stainless Steel Bolting	Loss of Preload/Stress Relaxation	PWR Vessel Internals	Not Applicable
Core Barrel Assembly (outlet nozzle)	Stainless Steel	Changes in Dimensions/Void Swelling	PWR Vessel Internals	Not Applicable
	Stainless Steel	Cracking/Stress Corrosion Cracking, Irradiation-Assisted Stress Corrosion Cracking	PWR Vessel Internals	Not Applicable
	Stainless Steel	Cracking/Stress Corrosion Cracking, Irradiation-Assisted Stress Corrosion Cracking	Water Chemistry	Not Applicable
	Stainless Steel	Loss of Fracture Toughness/Neutron Irradiation Embrittlement, Void Swelling	PWR Vessel Internals	Not Applicable
	Stainless Steel	Loss of Material/Pitting and Crevice Corrosion	Water Chemistry	Not Applicable

Table B-1 Summary of SGS LRA Table 3.1.2-3 – Reactor Vessel Internals Summary of Aging Management Evaluation (cont.)

Component Type	Material	Aging Effect Requiring Management	Aging Management Programs	Comments
Core Barrel Assembly (spray nozzles)	Stainless Steel	Changes in Dimensions/Void Swelling	PWR Vessel Internals	Not Applicable
	Stainless Steel	Cracking/Stress Corrosion Cracking, Irradiation-Assisted Stress Corrosion Cracking	PWR Vessel Internals	Not Applicable
	Stainless Steel	Cracking/Stress Corrosion Cracking, Irradiation-Assisted Stress Corrosion Cracking	Water Chemistry	Not Applicable
	Stainless Steel	Loss of Fracture Toughness/Neutron Irradiation Embrittlement, Void Swelling	PWR Vessel Internals	Not Applicable
	Stainless Steel	Loss of Material/Pitting and Crevice Corrosion	Water Chemistry	Not Applicable
Fuel Assembly (Short-lived)	Not Applicable	Not Applicable	Not Applicable	Not Applicable
Lower Internal Assembly (axial flexures: thermal shield to core barrel)	Stainless Steel	Changes in Dimensions/Void Swelling	PWR Vessel Internals	Not Applicable
	Stainless Steel	Cracking/Stress Corrosion Cracking, Irradiation-Assisted Stress Corrosion Cracking	PWR Vessel Internals	Not Applicable
	Stainless Steel	Cracking/Stress Corrosion Cracking, Irradiation-Assisted Stress Corrosion Cracking	Water Chemistry	Not Applicable
	Stainless Steel	Loss of Fracture Toughness/Neutron Irradiation Embrittlement, Void Swelling	PWR Vessel Internals	Not Applicable
	Stainless Steel	Loss of Material/Pitting and Crevice Corrosion	Water Chemistry	Not Applicable
	Stainless Steel	Loss of Material/Wear	ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD	Not Applicable

Table B-1 Summary of SGS LRA Table 3.1.2-3 – Reactor Vessel Internals Summary of Aging Management Evaluation

(cont.)

Component Type	Material	Aging Effect Requiring Management	Aging Management Programs	Comments
Lower Internal Assembly (clevis block bolts)	Nickel Alloy	Changes in Dimensions/Void Swelling	PWR Vessel Internals	Not Applicable
	Nickel Alloy	Cracking/Stress Corrosion Cracking, Irradiation-Assisted Stress Corrosion Cracking	PWR Vessel Internals	Not Applicable
	Nickel Alloy	Cracking/Stress Corrosion Cracking, Irradiation-Assisted Stress Corrosion Cracking	Water Chemistry	Not Applicable
	Nickel Alloy	Loss of Fracture Toughness/Neutron Irradiation Embrittlement, Void Swelling	PWR Vessel Internals	Not Applicable
	Nickel Alloy	Loss of Material/Pitting and Crevice Corrosion	Water Chemistry	Not Applicable
	Nickel Alloy	Loss of Preload/Stress Relaxation	PWR Vessel Internals	Not Applicable
Lower Internal Assembly (clevis block lock keys)	Nickel Alloy	Changes in Dimensions/Void Swelling	PWR Vessel Internals	Not Applicable
	Nickel Alloy	Cracking/Stress Corrosion Cracking, Irradiation-Assisted Stress Corrosion Cracking	PWR Vessel Internals	Not Applicable
	Nickel Alloy	Cracking/Stress Corrosion Cracking, Irradiation-Assisted Stress Corrosion Cracking	Water Chemistry	Not Applicable
	Nickel Alloy	Loss of Material/Pitting and Crevice Corrosion	Water Chemistry	Not Applicable
	Nickel Alloy	Loss of Material/Wear	ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD	Not Applicable

Table B-1 Summary of SGS LRA Table 3.1.2-3 – Reactor Vessel Internals Summary of Aging Management Evaluation

(cont.)

Component Type	Material	Aging Effect Requiring Management	Aging Management Programs	Comments
Lower Internal Assembly (clevis blocks and inserts)	Nickel Alloy	Changes in Dimensions/Void Swelling	PWR Vessel Internals	Not Applicable
	Nickel Alloy	Cracking/Stress Corrosion Cracking, Irradiation-Assisted Stress Corrosion Cracking	PWR Vessel Internals	Not Applicable
	Nickel Alloy	Cracking/Stress Corrosion Cracking, Irradiation-Assisted Stress Corrosion Cracking	Water Chemistry	Not Applicable
	Nickel Alloy	Loss of Material/Pitting and Crevice Corrosion	Water Chemistry	Not Applicable
	Nickel Alloy	Loss of Material/Wear	ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD	Not Applicable
Lower Internal Assembly (core support dome)	Stainless Steel	Changes in Dimensions/Void Swelling	PWR Vessel Internals	Note 1
	Stainless Steel	Cracking/Stress Corrosion Cracking, Irradiation-Assisted Stress Corrosion Cracking	PWR Vessel Internals	Note 1
	Stainless Steel	Cracking/Stress Corrosion Cracking, Irradiation-Assisted Stress Corrosion Cracking	Water Chemistry	Note 1
	Stainless Steel	Loss of Fracture Toughness/Neutron Irradiation Embrittlement, Void Swelling	PWR Vessel Internals	Note 1
	Stainless Steel	Loss of Material/Pitting and Crevice Corrosion	Water Chemistry	Note 1

Table B-1 Summary of SGS LRA Table 3.1.2-3 – Reactor Vessel Internals Summary of Aging Management Evaluation

(cont.)

Component Type	Material	Aging Effect Requiring Management	Aging Management Programs	Comments
Lower Internal Assembly (core support, incl'g core support lugs, columns and sleeves)	Nickel Alloy	Changes in Dimensions/Void Swelling	PWR Vessel Internals	Note 4
	Nickel Alloy	Cracking/Stress Corrosion Cracking, Irradiation-Assisted Stress Corrosion Cracking	PWR Vessel Internals	Note 4
	Nickel Alloy	Cracking/Stress Corrosion Cracking, Irradiation-Assisted Stress Corrosion Cracking	Water Chemistry	Note 4
	Nickel Alloy	Cumulative Fatigue Damage/Fatigue	TLAA	Note 4
	Nickel Alloy	Loss of Material/Pitting and Crevice Corrosion	Water Chemistry	Note 4
Lower Internal Assembly (core support, incl'g core support lugs, columns and sleeves)	Stainless Steel	Changes in Dimensions/Void Swelling	PWR Vessel Internals	Not Applicable
	Stainless Steel	Cracking/Stress Corrosion Cracking, Irradiation-Assisted Stress Corrosion Cracking	PWR Vessel Internals	Not Applicable
	Stainless Steel	Cracking/Stress Corrosion Cracking, Irradiation-Assisted Stress Corrosion Cracking	Water Chemistry	Not Applicable
	Stainless Steel	Loss of Fracture Toughness/Neutron Irradiation Embrittlement, Void Swelling	PWR Vessel Internals	Not Applicable
	Stainless Steel	Loss of Material/Pitting and Crevice Corrosion	Water Chemistry	Not Applicable

Table B-1 Summary of SGS LRA Table 3.1.2-3 – Reactor Vessel Internals Summary of Aging Management Evaluation

(cont.)

Component Type	Material	Aging Effect Requiring Management	Aging Management Programs	Comments
Lower Internal Assembly (flow distributor (diffuser) plate)	Stainless Steel	Changes in Dimensions/Void Swelling	PWR Vessel Internals	Not Applicable
	Stainless Steel	Cracking/Stress Corrosion Cracking, Irradiation-Assisted Stress Corrosion Cracking	PWR Vessel Internals	Not Applicable
	Stainless Steel	Cracking/Stress Corrosion Cracking, Irradiation-Assisted Stress Corrosion Cracking	Water Chemistry	Not Applicable
	Stainless Steel	Loss of Fracture Toughness/Neutron Irradiation Embrittlement, Void Swelling	PWR Vessel Internals	Not Applicable
	Stainless Steel	Loss of Material/Pitting and Crevice Corrosion	Water Chemistry	Not Applicable
Lower Internal Assembly (fuel assembly locating pin bolts)	Stainless Steel Bolting	Changes in Dimensions/Void Swelling	PWR Vessel Internals	Not Applicable
	Stainless Steel Bolting	Cracking/Stress Corrosion Cracking, Irradiation-Assisted Stress Corrosion Cracking	PWR Vessel Internals	Not Applicable
	Stainless Steel Bolting	Cracking/Stress Corrosion Cracking, Irradiation-Assisted Stress Corrosion Cracking	Water Chemistry	Not Applicable
	Stainless Steel Bolting	Loss of Fracture Toughness/Neutron Irradiation Embrittlement, Void Swelling	PWR Vessel Internals	Not Applicable
	Stainless Steel Bolting	Loss of Material/Pitting and Crevice Corrosion	Water Chemistry	Not Applicable
	Stainless Steel Bolting	Loss of Preload/Stress Relaxation	PWR Vessel Internals	Not Applicable

Table B-1 Summary of SGS LRA Table 3.1.2-3 – Reactor Vessel Internals Summary of Aging Management Evaluation

(cont.)

Component Type	Material	Aging Effect Requiring Management	Aging Management Programs	Comments
Lower Internal Assembly (fuel assembly locating pins and lockcaps)	Stainless Steel	Changes in Dimensions/Void Swelling	PWR Vessel Internals	Not Applicable
	Stainless Steel	Cracking/Stress Corrosion Cracking, Irradiation-Assisted Stress Corrosion Cracking	PWR Vessel Internals	Not Applicable
	Stainless Steel	Cracking/Stress Corrosion Cracking, Irradiation-Assisted Stress Corrosion Cracking	Water Chemistry	Not Applicable
	Stainless Steel	Loss of Fracture Toughness/Neutron Irradiation Embrittlement, Void Swelling	PWR Vessel Internals	Not Applicable
	Stainless Steel	Loss of Material/Pitting and Crevice Corrosion	Water Chemistry	Not Applicable
Lower Internal Assembly (inserts for clevis blocks, incl'g lock bars and dowels)	Nickel Alloy	Changes in Dimensions/Void Swelling	PWR Vessel Internals	Not Applicable
	Nickel Alloy	Cracking/Stress Corrosion Cracking, Irradiation-Assisted Stress Corrosion Cracking	PWR Vessel Internals	Not Applicable
	Nickel Alloy	Cracking/Stress Corrosion Cracking, Irradiation-Assisted Stress Corrosion Cracking	Water Chemistry	Not Applicable
	Nickel Alloy	Loss of Material/Pitting and Crevice Corrosion	Water Chemistry	Not Applicable
	Nickel Alloy	Loss of Material/Wear	ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD	Not Applicable

Table B-1 Summary of SGS LRA Table 3.1.2-3 – Reactor Vessel Internals Summary of Aging Management Evaluation

(cont.)

Component Type	Material	Aging Effect Requiring Management	Aging Management Programs	Comments
Lower Internal Assembly (irradiation sample access plugs)	Stainless Steel	Changes in Dimensions/Void Swelling	PWR Vessel Internals	Not Applicable
	Stainless Steel	Cracking/Stress Corrosion Cracking, Irradiation-Assisted Stress Corrosion Cracking	PWR Vessel Internals	Not Applicable
	Stainless Steel	Cracking/Stress Corrosion Cracking, Irradiation-Assisted Stress Corrosion Cracking	Water Chemistry	Not Applicable
	Stainless Steel	Loss of Fracture Toughness/Neutron Irradiation Embrittlement, Void Swelling	PWR Vessel Internals	Not Applicable
	Stainless Steel	Loss of Material/Pitting and Crevice Corrosion	Water Chemistry	Not Applicable
	Stainless Steel	Loss of Preload/Stress Relaxation	PWR Vessel Internals	Not Applicable
Lower Internal Assembly (irradiation sample guide bolts)	Stainless Steel Bolting	Changes in Dimensions/Void Swelling	PWR Vessel Internals	Note 2
	Stainless Steel Bolting	Cracking/Stress Corrosion Cracking, Irradiation-Assisted Stress Corrosion Cracking	PWR Vessel Internals	Note 2
	Stainless Steel Bolting	Cracking/Stress Corrosion Cracking, Irradiation-Assisted Stress Corrosion Cracking	Water Chemistry	Note 2
	Stainless Steel Bolting	Loss of Fracture Toughness/Neutron Irradiation Embrittlement, Void Swelling	PWR Vessel Internals	Note 2
	Stainless Steel Bolting	Loss of Material/Pitting and Crevice Corrosion	Water Chemistry	Note 2
	Stainless Steel Bolting	Loss of Preload/Stress Relaxation	PWR Vessel Internals	Note 2

Table B-1 Summary of SGS LRA Table 3.1.2-3 – Reactor Vessel Internals Summary of Aging Management Evaluation

(cont.)

Component Type	Material	Aging Effect Requiring Management	Aging Management Programs	Comments
Lower Internal Assembly (irradiation sample guide lock caps)	Stainless Steel	Changes in Dimensions/Void Swelling	PWR Vessel Internals	Note 2
	Stainless Steel	Cracking/Stress Corrosion Cracking, Irradiation-Assisted Stress Corrosion Cracking	PWR Vessel Internals	Note 2
	Stainless Steel	Cracking/Stress Corrosion Cracking, Irradiation-Assisted Stress Corrosion Cracking	Water Chemistry	Note 2
	Stainless Steel	Loss of Material/Pitting and Crevice Corrosion	Water Chemistry	Note 2
	Stainless Steel	Loss of Material/Wear	ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD	Note 2
Lower Internal Assembly (irradiation sample guide)	Not Applicable	Not Applicable	Not Applicable	Not Applicable
Lower Internal Assembly (lower core plate)	Stainless Steel	Changes in Dimensions/Void Swelling	PWR Vessel Internals	Not Applicable
	Stainless Steel	Cracking/Stress Corrosion Cracking, Irradiation-Assisted Stress Corrosion Cracking	PWR Vessel Internals	Not Applicable
	Stainless Steel	Cracking/Stress Corrosion Cracking, Irradiation-Assisted Stress Corrosion Cracking	Water Chemistry	Not Applicable
	Stainless Steel	Cumulative Fatigue Damage/Fatigue	TLAA	Not Applicable
	Stainless Steel	Loss of Fracture Toughness/Neutron Irradiation Embrittlement, Void Swelling	PWR Vessel Internals	Not Applicable
	Stainless Steel	Loss of Material/Pitting and Crevice Corrosion	Water Chemistry	Not Applicable

Table B-1 Summary of SGS LRA Table 3.1.2-3 – Reactor Vessel Internals Summary of Aging Management Evaluation

(cont.)

Component Type	Material	Aging Effect Requiring Management	Aging Management Programs	Comments
Lower Internal Assembly (lower core support energy absorbers)	Stainless Steel	Changes in Dimensions/Void Swelling	PWR Vessel Internals	Not Applicable
	Stainless Steel	Cracking/Stress Corrosion Cracking, Irradiation-Assisted Stress Corrosion Cracking	PWR Vessel Internals	Not Applicable
	Stainless Steel	Cracking/Stress Corrosion Cracking, Irradiation-Assisted Stress Corrosion Cracking	Water Chemistry	Not Applicable
	Stainless Steel	Loss of Fracture Toughness/Neutron Irradiation Embrittlement, Void Swelling	PWR Vessel Internals	Not Applicable
	Stainless Steel	Loss of Material/Pitting and Crevice Corrosion	Water Chemistry	Not Applicable
Lower Internal Assembly (lower core support guide post and housing)	Stainless Steel	Changes in Dimensions/Void Swelling	PWR Vessel Internals	Not Applicable
	Stainless Steel	Cracking/Stress Corrosion Cracking, Irradiation-Assisted Stress Corrosion Cracking	PWR Vessel Internals	Not Applicable
	Stainless Steel	Cracking/Stress Corrosion Cracking, Irradiation-Assisted Stress Corrosion Cracking	Water Chemistry	Not Applicable
	Stainless Steel	Loss of Fracture Toughness/Neutron Irradiation Embrittlement, Void Swelling	PWR Vessel Internals	Not Applicable
	Stainless Steel	Loss of Material/Pitting and Crevice Corrosion	Water Chemistry	Not Applicable

Table B-1 Summary of SGS LRA Table 3.1.2-3 – Reactor Vessel Internals Summary of Aging Management Evaluation

(cont.)

Component Type	Material	Aging Effect Requiring Management	Aging Management Programs	Comments
Lower Internal Assembly (lower core support ring)	Stainless Steel	Changes in Dimensions/Void Swelling	PWR Vessel Internals	Not Applicable
	Stainless Steel	Cracking/Stress Corrosion Cracking, Irradiation-Assisted Stress Corrosion Cracking	PWR Vessel Internals	Not Applicable
	Stainless Steel	Cracking/Stress Corrosion Cracking, Irradiation-Assisted Stress Corrosion Cracking	Water Chemistry	Not Applicable
	Stainless Steel	Loss of Fracture Toughness/Neutron Irradiation Embrittlement, Void Swelling	PWR Vessel Internals	Not Applicable
	Stainless Steel	Loss of Material/Pitting and Crevice Corrosion	Water Chemistry	Not Applicable
Lower Internal Assembly (lower radial support keys)	Stainless Steel	Changes in Dimensions/Void Swelling	PWR Vessel Internals	Not Applicable
	Stainless Steel	Cracking/Stress Corrosion Cracking, Irradiation-Assisted Stress Corrosion Cracking	PWR Vessel Internals	Not Applicable
	Stainless Steel	Cracking/Stress Corrosion Cracking, Irradiation-Assisted Stress Corrosion Cracking	Water Chemistry	Not Applicable
	Stainless Steel	Loss of Material/Pitting and Crevice Corrosion	Water Chemistry	Not Applicable
	Stainless Steel	Loss of Material/Wear	ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD	Not Applicable

Table B-1 Summary of SGS LRA Table 3.1.2-3 – Reactor Vessel Internals Summary of Aging Management Evaluation (cont.)

Component Type	Material	Aging Effect Requiring Management	Aging Management Programs	Comments
Lower Internal Assembly (lower support base bolt lock keys)	Stainless Steel	Changes in Dimensions/Void Swelling	PWR Vessel Internals	Not Applicable
	Stainless Steel	Cracking/Stress Corrosion Cracking, Irradiation-Assisted Stress Corrosion Cracking	PWR Vessel Internals	Not Applicable
	Stainless Steel	Cracking/Stress Corrosion Cracking, Irradiation-Assisted Stress Corrosion Cracking	Water Chemistry	Not Applicable
	Stainless Steel	Loss of Material/Pitting and Crevice Corrosion	Water Chemistry	Not Applicable
	Stainless Steel	Loss of Material/Wear	ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD	Not Applicable
Lower Internal Assembly (lower support base bolts)	Stainless Steel Bolting	Changes in Dimensions/Void Swelling	PWR Vessel Internals	Not Applicable
	Stainless Steel Bolting	Cracking/Stress Corrosion Cracking, Irradiation-Assisted Stress Corrosion Cracking	PWR Vessel Internals	Not Applicable
	Stainless Steel Bolting	Cracking/Stress Corrosion Cracking, Irradiation-Assisted Stress Corrosion Cracking	Water Chemistry	Not Applicable
	Stainless Steel Bolting	Loss of Fracture Toughness/Neutron Irradiation Embrittlement, Void Swelling	PWR Vessel Internals	Not Applicable
	Stainless Steel Bolting	Loss of Material/Pitting and Crevice Corrosion	Water Chemistry	Not Applicable
	Stainless Steel Bolting	Loss of Preload/Stress Relaxation	PWR Vessel Internals	Not Applicable

Table B-1 (cont.) Summary of SGS LRA Table 3.1.2-3 – Reactor Vessel Internals Summary of Aging Management Evaluation

Component Type	Material	Aging Effect Requiring Management	Aging Management Programs	Comments
Lower Internal Assembly (lower support column bolts)	Stainless Steel Bolting	Changes in Dimensions/Void Swelling	PWR Vessel Internals	Not Applicable
	Stainless Steel Bolting	Cracking/Stress Corrosion Cracking, Irradiation-Assisted Stress Corrosion Cracking	PWR Vessel Internals	Not Applicable
	Stainless Steel Bolting	Cracking/Stress Corrosion Cracking, Irradiation-Assisted Stress Corrosion Cracking	Water Chemistry	Not Applicable
	Stainless Steel Bolting	Loss of Fracture Toughness/Neutron Irradiation Embrittlement, Void Swelling	PWR Vessel Internals	Not Applicable
	Stainless Steel Bolting	Loss of Preload/Stress Relaxation	PWR Vessel Internals	Not Applicable
Lower Internal Assembly (lower support column nuts)	Stainless Steel	Changes in Dimensions/Void Swelling	PWR Vessel Internals	Not Applicable
	Stainless Steel	Cracking/Stress Corrosion Cracking, Irradiation-Assisted Stress Corrosion Cracking	PWR Vessel Internals	Not Applicable
	Stainless Steel	Cracking/Stress Corrosion Cracking, Irradiation-Assisted Stress Corrosion Cracking	Water Chemistry	Not Applicable
	Stainless Steel	Loss of Fracture Toughness/Neutron Irradiation Embrittlement, Void Swelling	PWR Vessel Internals	Not Applicable
	Stainless Steel	Loss of Material/Pitting and Crevice Corrosion	Water Chemistry	Not Applicable

Table B-1 Summary of SGS LRA Table 3.1.2-3 – Reactor Vessel Internals Summary of Aging Management Evaluation

(cont.)

Component Type	Material	Aging Effect Requiring Management	Aging Management Programs	Comments
Lower Internal Assembly (lower support lock keys)	Stainless Steel	Changes in Dimensions/Void Swelling	PWR Vessel Internals	Note 7
	Stainless Steel	Cracking/Stress Corrosion Cracking, Irradiation-Assisted Stress Corrosion Cracking	PWR Vessel Internals	Note 7
	Stainless Steel	Cracking/Stress Corrosion Cracking, Irradiation-Assisted Stress Corrosion Cracking	Water Chemistry	Note 7
	Stainless Steel	Loss of Material/Pitting and Crevice Corrosion	Water Chemistry	Note 7
	Stainless Steel	Loss of Material/Wear	ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD	Note 7
RCCA Guide Tube Assemblies (bolts)	Stainless Steel Bolting	Changes in Dimensions/Void Swelling	PWR Vessel Internals	Not Applicable
	Stainless Steel Bolting	Cracking/Stress Corrosion Cracking, Irradiation-Assisted Stress Corrosion Cracking	PWR Vessel Internals	Not Applicable
	Stainless Steel Bolting	Cracking/Stress Corrosion Cracking, Irradiation-Assisted Stress Corrosion Cracking	Water Chemistry	Not Applicable
	Stainless Steel Bolting	Loss of Material/Pitting and Crevice Corrosion	Water Chemistry	Not Applicable
	Stainless Steel Bolting	Loss of Preload/Stress Relaxation	PWR Vessel Internals	Not Applicable

Table B-1 Summary of SGS LRA Table 3.1.2-3 – Reactor Vessel Internals Summary of Aging Management Evaluation

(cont.)

Component Type	Material	Aging Effect Requiring Management	Aging Management Programs	Comments
RCCA Guide Tube Assemblies (enclosures)	Stainless Steel	Changes in Dimensions/Void Swelling	PWR Vessel Internals	Not Applicable
	Stainless Steel	Cracking/Stress Corrosion Cracking, Irradiation-Assisted Stress Corrosion Cracking	PWR Vessel Internals	Not Applicable
	Stainless Steel	Cracking/Stress Corrosion Cracking, Irradiation-Assisted Stress Corrosion Cracking	Water Chemistry	Not Applicable
	Stainless Steel	Loss of Material/Pitting and Crevice Corrosion	Water Chemistry	Not Applicable
RCCA guide tube assemblies (flexures; inserts)	Nickel Alloy	Changes in Dimensions/Void Swelling	PWR Vessel Internals	Note 2
	Nickel Alloy	Cracking/Stress Corrosion Cracking, Irradiation-Assisted Stress Corrosion Cracking	PWR Vessel Internals	Note 2
	Nickel Alloy	Cracking/Stress Corrosion Cracking, Irradiation-Assisted Stress Corrosion Cracking	Water Chemistry	Note 2
	Nickel Alloy	Loss of Material/Pitting and Crevice Corrosion	Water Chemistry	Note 2
	Nickel Alloy	Loss of Material/Wear	ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD	Note 2
RCCA guide tube assemblies (guide pins in tubes)	Stainless Steel	Changes in Dimensions/Void Swelling	PWR Vessel Internals	Not Applicable
	Stainless Steel	Cracking/Stress Corrosion Cracking, Irradiation-Assisted Stress Corrosion Cracking	PWR Vessel Internals	Not Applicable
	Stainless Steel	Cracking/Stress Corrosion Cracking, Irradiation-Assisted Stress Corrosion Cracking	Water Chemistry	Not Applicable
	Stainless Steel	Loss of Material/Pitting and Crevice Corrosion	Water Chemistry	Not Applicable

Table B-1 (cont.) Summary of SGS LRA Table 3.1.2-3 – Reactor Vessel Internals Summary of Aging Management Evaluation

Component Type	Material	Aging Effect Requiring Management	Aging Management Programs	Comments
RCCA guide tube assemblies (lock bars)	Stainless Steel	Changes in Dimensions/Void Swelling	PWR Vessel Internals	Note 2
	Stainless Steel	Cracking/Stress Corrosion Cracking, Irradiation-Assisted Stress Corrosion Cracking	PWR Vessel Internals	Note 2
	Stainless Steel	Cracking/Stress Corrosion Cracking, Irradiation-Assisted Stress Corrosion Cracking	Water Chemistry	Note 2
	Stainless Steel	Loss of Material/Pitting and Crevice Corrosion	Water Chemistry	Note 2
RCCA guide tube assemblies (lower flanges)	Cast Austenitic Stainless Steel (CASS)	Changes in Dimensions/Void Swelling	PWR Vessel Internals	Note 5
	Cast Austenitic Stainless Steel (CASS)	Cracking/Stress Corrosion Cracking, Irradiation-Assisted Stress Corrosion Cracking	PWR Vessel Internals	Note 5
	Cast Austenitic Stainless Steel (CASS)	Cracking/Stress Corrosion Cracking, Irradiation-Assisted Stress Corrosion Cracking	Water Chemistry	Note 5
	Cast Austenitic Stainless Steel (CASS)	Loss of Material/Pitting and Crevice Corrosion	Water Chemistry	Note 5
RCCA guide tube assemblies (pins, anti-rotation studs, and nuts)	Stainless Steel	Changes in Dimensions/Void Swelling	PWR Vessel Internals	Note 2
	Stainless Steel	Cracking/Stress Corrosion Cracking, Irradiation-Assisted Stress Corrosion Cracking	PWR Vessel Internals	Note 2
	Stainless Steel	Cracking/Stress Corrosion Cracking, Irradiation-Assisted Stress Corrosion Cracking	Water Chemistry	Note 2
	Stainless Steel	Loss of Material/Pitting and Crevice Corrosion	Water Chemistry	Note 2

Table B-1 Summary of SGS LRA Table 3.1.2-3 – Reactor Vessel Internals Summary of Aging Management Evaluation

(cont.)

Component Type	Material	Aging Effect Requiring Management	Aging Management Programs	Comments
RCCA guide tube assemblies (sheaths)	Stainless Steel	Changes in Dimensions/Void Swelling	PWR Vessel Internals	Not Applicable
	Stainless Steel	Cracking/Stress Corrosion Cracking, Irradiation-Assisted Stress Corrosion Cracking	PWR Vessel Internals	Not Applicable
	Stainless Steel	Cracking/Stress Corrosion Cracking, Irradiation-Assisted Stress Corrosion Cracking	Water Chemistry	Not Applicable
	Stainless Steel	Loss of Material/Pitting and Crevice Corrosion	Water Chemistry	Not Applicable
RCCA guide tube assemblies (support pin cover plate)	Stainless Steel	Changes in Dimensions/Void Swelling	PWR Vessel Internals	Note 2
	Stainless Steel	Cracking/Stress Corrosion Cracking, Irradiation-Assisted Stress Corrosion Cracking	PWR Vessel Internals	Note 2
	Stainless Steel	Cracking/Stress Corrosion Cracking, Irradiation-Assisted Stress Corrosion Cracking	Water Chemistry	Note 2
	Stainless Steel	Loss of Material/Pitting and Crevice Corrosion	Water Chemistry	Note 2
RCCA guide tube assemblies (support pin fasteners and nuts)	Stainless Steel	Changes in Dimensions/Void Swelling	PWR Vessel Internals	Not Applicable
	Stainless Steel	Cracking/Stress Corrosion Cracking, Irradiation-Assisted Stress Corrosion Cracking	PWR Vessel Internals	Not Applicable
	Stainless Steel	Cracking/Stress Corrosion Cracking, Irradiation-Assisted Stress Corrosion Cracking	Water Chemistry	Not Applicable
	Stainless Steel	Loss of Material/Pitting and Crevice Corrosion	Water Chemistry	Not Applicable

Table B-1 Summary of SGS LRA Table 3.1.2-3 – Reactor Vessel Internals Summary of Aging Management Evaluation (cont.)

Component Type	Material	Aging Effect Requiring Management	Aging Management Programs	Comments
RCCA guide tube assemblies (tubes, housing plates, and guide plates)	Stainless Steel	Changes in Dimensions/Void Swelling	PWR Vessel Internals	Note 5
	Stainless Steel	Cracking/Stress Corrosion Cracking, Irradiation-Assisted Stress Corrosion Cracking	PWR Vessel Internals	Note 5
	Stainless Steel	Cracking/Stress Corrosion Cracking, Irradiation-Assisted Stress Corrosion Cracking	Water Chemistry	Note 5
	Stainless Steel	Loss of Material/Pitting and Crevice Corrosion	Water Chemistry	Note 5
RCCA guide tube assemblies (upper guide tube)	Stainless Steel	Changes in Dimensions/Void Swelling	PWR Vessel Internals	Not Applicable
	Stainless Steel	Cracking/Stress Corrosion Cracking, Irradiation-Assisted Stress Corrosion Cracking	PWR Vessel Internals	Not Applicable
	Stainless Steel	Cracking/Stress Corrosion Cracking, Irradiation-Assisted Stress Corrosion Cracking	Water Chemistry	Not Applicable
	Stainless Steel	Loss of Material/Pitting and Crevice Corrosion	Water Chemistry	Not Applicable
Reactor Vessel Internals (core support locking nut)	Stainless Steel	Changes in Dimensions/Void Swelling	PWR Vessel Internals	Not Applicable
	Stainless Steel	Cracking/Stress Corrosion Cracking, Irradiation-Assisted Stress Corrosion Cracking	PWR Vessel Internals	Not Applicable
	Stainless Steel	Cracking/Stress Corrosion Cracking, Irradiation-Assisted Stress Corrosion Cracking	Water Chemistry	Not Applicable
	Stainless Steel	Loss of Fracture Toughness/Neutron Irradiation Embrittlement, Void Swelling	PWR Vessel Internals	Not Applicable
	Stainless Steel	Loss of Material/Pitting and Crevice Corrosion	Water Chemistry	Not Applicable

Table B-1 Summary of SGS LRA Table 3.1.2-3 – Reactor Vessel Internals Summary of Aging Management Evaluation

(cont.)

Component Type	Material	Aging Effect Requiring Management	Aging Management Programs	Comments
Reactor Vessel Internals (flux thimbles - tubes)	Stainless Steel	Changes in Dimensions/Void Swelling	PWR Vessel Internals	Note 6
	Stainless Steel	Cracking/Stress Corrosion Cracking, Irradiation-Assisted Stress Corrosion Cracking	PWR Vessel Internals	Note 6
	Stainless Steel	Cracking/Stress Corrosion Cracking, Irradiation-Assisted Stress Corrosion Cracking	Water Chemistry	Note 6
	Stainless Steel	Loss of Material/Pitting and Crevice Corrosion	Water Chemistry	Note 6
	Stainless Steel	Loss of Material/Wear	Flux Thimble Tube Inspection	Note 6
Reactor Vessel Internals (incore guide cruciforms)	Cast Austenitic Stainless Steel (CASS)	Changes in Dimensions/Void Swelling	PWR Vessel Internals	Note 3
	Cast Austenitic Stainless Steel (CASS)	Cracking/Stress Corrosion Cracking, Irradiation-Assisted Stress Corrosion Cracking	PWR Vessel Internals	Note 3
	Cast Austenitic Stainless Steel (CASS)	Cracking/Stress Corrosion Cracking, Irradiation-Assisted Stress Corrosion Cracking	Water Chemistry	Note 3
	Cast Austenitic Stainless Steel (CASS)	Loss of Fracture Toughness/Neutron Irradiation Embrittlement, Void Swelling	PWR Vessel Internals	Note 3
	Cast Austenitic Stainless Steel (CASS)	Loss of Fracture Toughness/Thermal Aging and Neutron Irradiation Embrittlement	PWR Vessel Internals	Note 3
	Cast Austenitic Stainless Steel (CASS)	Loss of Material/Pitting and Crevice Corrosion	Water Chemistry	Note 3

Table B-1 Summary of SGS LRA Table 3.1.2-3 – Reactor Vessel Internals Summary of Aging Management Evaluation

(cont.)

Component Type	Material	Aging Effect Requiring Management	Aging Management Programs	Comments
Reactor Vessel Internals (incore guide tube column bodies)	Stainless Steel	Changes in Dimensions/Void Swelling	PWR Vessel Internals	Not Applicable
	Stainless Steel	Cracking/Stress Corrosion Cracking, Irradiation-Assisted Stress Corrosion Cracking	PWR Vessel Internals	Not Applicable
	Stainless Steel	Cracking/Stress Corrosion Cracking, Irradiation-Assisted Stress Corrosion Cracking	Water Chemistry	Not Applicable
	Stainless Steel	Loss of Fracture Toughness/Neutron Irradiation Embrittlement, Void Swelling	PWR Vessel Internals	Not Applicable
	Stainless Steel	Loss of Material/Pitting and Crevice Corrosion	Water Chemistry	Not Applicable
Reactor Vessel Internals (incore guide tube extensions)	Stainless Steel	Changes in Dimensions/Void Swelling	PWR Vessel Internals	Not Applicable
	Stainless Steel	Cracking/Stress Corrosion Cracking, Irradiation-Assisted Stress Corrosion Cracking	PWR Vessel Internals	Not Applicable
	Stainless Steel	Cracking/Stress Corrosion Cracking, Irradiation-Assisted Stress Corrosion Cracking	Water Chemistry	Not Applicable
	Stainless Steel	Loss of Fracture Toughness/Neutron Irradiation Embrittlement, Void Swelling	PWR Vessel Internals	Not Applicable
	Stainless Steel	Loss of Material/Pitting and Crevice Corrosion	Water Chemistry	Not Applicable

Table B-1 Summary of SGS LRA Table 3.1.2-3 – Reactor Vessel Internals Summary of Aging Management Evaluation

(cont.)

Component Type	Material	Aging Effect Requiring Management	Aging Management Programs	Comments
Reactor Vessel Internals (incore instrument guide extension bolt lock caps)	Stainless Steel	Changes in Dimensions/Void Swelling	PWR Vessel Internals	Not Applicable
	Stainless Steel	Cracking/Stress Corrosion Cracking, Irradiation-Assisted Stress Corrosion Cracking	PWR Vessel Internals	Not Applicable
	Stainless Steel	Cracking/Stress Corrosion Cracking, Irradiation-Assisted Stress Corrosion Cracking	Water Chemistry	Not Applicable
	Stainless Steel	Loss of Fracture Toughness/Neutron Irradiation Embrittlement, Void Swelling	PWR Vessel Internals	Not Applicable
	Stainless Steel	Loss of Material/Pitting and Crevice Corrosion	Water Chemistry	Not Applicable
Reactor Vessel Internals (incore instrument guide extension bolts)	Stainless Steel Bolting	Changes in Dimensions/Void Swelling	PWR Vessel Internals	Not Applicable
	Stainless Steel Bolting	Cracking/Stress Corrosion Cracking, Irradiation-Assisted Stress Corrosion Cracking	PWR Vessel Internals	Not Applicable
	Stainless Steel Bolting	Cracking/Stress Corrosion Cracking, Irradiation-Assisted Stress Corrosion Cracking	Water Chemistry	Not Applicable
	Stainless Steel Bolting	Loss of Fracture Toughness/Neutron Irradiation Embrittlement, Void Swelling	PWR Vessel Internals	Not Applicable
	Stainless Steel Bolting	Loss of Material/Pitting and Crevice Corrosion	Water Chemistry	Not Applicable
	Stainless Steel Bolting	Loss of Preload/Stress Relaxation	PWR Vessel Internals	Not Applicable

Table B-1 Summary of SGS LRA Table 3.1.2-3 – Reactor Vessel Internals Summary of Aging Management Evaluation

(cont.)

Component Type	Material	Aging Effect Requiring Management	Aging Management Programs	Comments
Reactor Vessel Internals (incore instrument guide extension collars)	Stainless Steel	Changes in Dimensions/Void Swelling	PWR Vessel Internals	Not Applicable
	Stainless Steel	Cracking/Stress Corrosion Cracking, Irradiation-Assisted Stress Corrosion Cracking	PWR Vessel Internals	Not Applicable
	Stainless Steel	Cracking/Stress Corrosion Cracking, Irradiation-Assisted Stress Corrosion Cracking	Water Chemistry	Not Applicable
	Stainless Steel	Loss of Fracture Toughness/Neutron Irradiation Embrittlement, Void Swelling	PWR Vessel Internals	Not Applicable
	Stainless Steel	Loss of Material/Pitting and Crevice Corrosion	Water Chemistry	Not Applicable
Reactor Vessel Internals (incore instrument guide extension nuts)	Stainless Steel	Changes in Dimensions/Void Swelling	PWR Vessel Internals	Not Applicable
	Stainless Steel	Cracking/Stress Corrosion Cracking, Irradiation-Assisted Stress Corrosion Cracking	PWR Vessel Internals	Not Applicable
	Stainless Steel	Cracking/Stress Corrosion Cracking, Irradiation-Assisted Stress Corrosion Cracking	Water Chemistry	Not Applicable
	Stainless Steel	Loss of Fracture Toughness/Neutron Irradiation Embrittlement, Void Swelling	PWR Vessel Internals	Not Applicable
	Stainless Steel	Loss of Material/Pitting and Crevice Corrosion	Water Chemistry	Not Applicable

Table B-1 (cont.) Summary of SGS LRA Table 3.1.2-3 – Reactor Vessel Internals Summary of Aging Management Evaluation

Component Type	Material	Aging Effect Requiring Management	Aging Management Programs	Comments
Reactor Vessel Internals (incore instrument guide tube extension bars)	Stainless Steel	Changes in Dimensions/Void Swelling	PWR Vessel Internals	Not Applicable
	Stainless Steel	Cracking/Stress Corrosion Cracking, Irradiation-Assisted Stress Corrosion Cracking	PWR Vessel Internals	Not Applicable
	Stainless Steel	Cracking/Stress Corrosion Cracking, Irradiation-Assisted Stress Corrosion Cracking	Water Chemistry	Not Applicable
	Stainless Steel	Loss of Fracture Toughness/Neutron Irradiation Embrittlement, Void Swelling	PWR Vessel Internals	Not Applicable
	Stainless Steel	Loss of Material/Pitting and Crevice Corrosion	Water Chemistry	Not Applicable
Reactor Vessel Internals (manway cover assembly)	Stainless Steel	Changes in Dimensions/Void Swelling	PWR Vessel Internals	Not Applicable
	Stainless Steel	Cracking/Stress Corrosion Cracking, Irradiation-Assisted Stress Corrosion Cracking	PWR Vessel Internals	Not Applicable
	Stainless Steel	Cracking/Stress Corrosion Cracking, Irradiation-Assisted Stress Corrosion Cracking	Water Chemistry	Not Applicable
	Stainless Steel	Loss of Fracture Toughness/Neutron Irradiation Embrittlement, Void Swelling	PWR Vessel Internals	Not Applicable
	Stainless Steel	Loss of Material/Pitting and Crevice Corrosion	Water Chemistry	Not Applicable

Table B-1 Summary of SGS LRA Table 3.1.2-3 – Reactor Vessel Internals Summary of Aging Management Evaluation (cont.)

Component Type	Material	Aging Effect Requiring Management	Aging Management Programs	Comments
Reactor Vessel Internals (secondary core support)	Stainless Steel	Changes in Dimensions/Void Swelling	PWR Vessel Internals	Not Applicable
	Stainless Steel	Cracking/Stress Corrosion Cracking, Irradiation-Assisted Stress Corrosion Cracking	PWR Vessel Internals	Not Applicable
	Stainless Steel	Cracking/Stress Corrosion Cracking, Irradiation-Assisted Stress Corrosion Cracking	Water Chemistry	Not Applicable
	Stainless Steel	Loss of Fracture Toughness/Neutron Irradiation Embrittlement, Void Swelling	PWR Vessel Internals	Not Applicable
	Stainless Steel	Loss of Material/Pitting and Crevice Corrosion	Water Chemistry	Not Applicable
Thermal Shield	Stainless Steel	Changes in Dimensions/Void Swelling	PWR Vessel Internals	Not Applicable
	Stainless Steel	Cracking/Stress Corrosion Cracking, Irradiation-Assisted Stress Corrosion Cracking	PWR Vessel Internals	Not Applicable
	Stainless Steel	Cracking/Stress Corrosion Cracking, Irradiation-Assisted Stress Corrosion Cracking	Water Chemistry	Not Applicable
	Stainless Steel	Loss of Fracture Toughness/Neutron Irradiation Embrittlement, Void Swelling	PWR Vessel Internals	Not Applicable
	Stainless Steel	Loss of Material/Pitting and Crevice Corrosion	Water Chemistry	Not Applicable

Table B-1 Summary of SGS LRA Table 3.1.2-3 – Reactor Vessel Internals Summary of Aging Management Evaluation

(cont.)

Component Type	Material	Aging Effect Requiring Management	Aging Management Programs	Comments
Thermal Shield (adjustment plugs)	Stainless Steel	Changes in Dimensions/Void Swelling	PWR Vessel Internals	Not Applicable
	Stainless Steel	Cracking/Stress Corrosion Cracking, Irradiation-Assisted Stress Corrosion Cracking	PWR Vessel Internals	Not Applicable
	Stainless Steel	Cracking/Stress Corrosion Cracking, Irradiation-Assisted Stress Corrosion Cracking	Water Chemistry	Not Applicable
	Stainless Steel	Loss of Fracture Toughness/Neutron Irradiation Embrittlement, Void Swelling	PWR Vessel Internals	Not Applicable
	Stainless Steel	Loss of Material/Pitting and Crevice Corrosion	Water Chemistry	Not Applicable
Thermal Shield (bolts and dowels)	Stainless Steel Bolting	Changes in Dimensions/Void Swelling	PWR Vessel Internals	Not Applicable
	Stainless Steel Bolting	Cracking/Stress Corrosion Cracking, Irradiation-Assisted Stress Corrosion Cracking	PWR Vessel Internals	Not Applicable
	Stainless Steel Bolting	Cracking/Stress Corrosion Cracking, Irradiation-Assisted Stress Corrosion Cracking	Water Chemistry	Not Applicable
	Stainless Steel Bolting	Loss of Fracture Toughness/Neutron Irradiation Embrittlement, Void Swelling	PWR Vessel Internals	Not Applicable
	Stainless Steel Bolting	Loss of Material/Pitting and Crevice Corrosion	Water Chemistry	Not Applicable
	Stainless Steel Bolting	Loss of Preload/Stress Relaxation	PWR Vessel Internals	Not Applicable

Table B-1 Summary of SGS LRA Table 3.1.2-3 – Reactor Vessel Internals Summary of Aging Management Evaluation

(cont.)

Component Type	Material	Aging Effect Requiring Management	Aging Management Programs	Comments
Upper Internals Assembly (beam and ribs bolts)	Stainless Steel Bolting	Changes in Dimensions/Void Swelling	PWR Vessel Internals	Not Applicable
	Stainless Steel Bolting	Cracking/Stress Corrosion Cracking, Irradiation-Assisted Stress Corrosion Cracking	PWR Vessel Internals	Not Applicable
	Stainless Steel Bolting	Cracking/Stress Corrosion Cracking, Irradiation-Assisted Stress Corrosion Cracking	Water Chemistry	Not Applicable
	Stainless Steel Bolting	Loss of Material/Pitting and Crevice Corrosion	Water Chemistry	Not Applicable
	Stainless Steel Bolting	Loss of Preload/Stress Relaxation	PWR Vessel Internals	Not Applicable
Upper Internals Assembly (beam and ribs lock keys)	Stainless Steel	Changes in Dimensions/Void Swelling	PWR Vessel Internals	Not Applicable
	Stainless Steel	Cracking/Stress Corrosion Cracking, Irradiation-Assisted Stress Corrosion Cracking	PWR Vessel Internals	Not Applicable
	Stainless Steel	Cracking/Stress Corrosion Cracking, Irradiation-Assisted Stress Corrosion Cracking	Water Chemistry	Not Applicable
	Stainless Steel	Loss of Material/Pitting and Crevice Corrosion	Water Chemistry	Not Applicable
	Stainless Steel	Loss of Material/Wear	ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD	Not Applicable

Table B-1 Summary of SGS LRA Table 3.1.2-3 – Reactor Vessel Internals Summary of Aging Management Evaluation

(cont.)

Component Type	Material	Aging Effect Requiring Management	Aging Management Programs	Comments
Upper Internals Assembly (capped top thermocouple columns)	Stainless Steel	Changes in Dimensions/Void Swelling	PWR Vessel Internals	Note 2
	Stainless Steel	Cracking/Stress Corrosion Cracking, Irradiation-Assisted Stress Corrosion Cracking	PWR Vessel Internals	Note 2
	Stainless Steel	Cracking/Stress Corrosion Cracking, Irradiation-Assisted Stress Corrosion Cracking	Water Chemistry	Note 2
	Stainless Steel	Loss of Material/Pitting and Crevice Corrosion	Water Chemistry	Note 2
Upper Internals Assembly (deep beam rib and stiffener, and ribs)	Stainless Steel	Changes in Dimensions/Void Swelling	PWR Vessel Internals	Not Applicable
	Stainless Steel	Cracking/Stress Corrosion Cracking, Irradiation-Assisted Stress Corrosion Cracking	PWR Vessel Internals	Not Applicable
	Stainless Steel	Cracking/Stress Corrosion Cracking, Irradiation-Assisted Stress Corrosion Cracking	Water Chemistry	Not Applicable
	Stainless Steel	Loss of Material/Pitting and Crevice Corrosion	Water Chemistry	Not Applicable
Upper Internals Assembly (fuel assembly locating pins)	Stainless Steel	Changes in Dimensions/Void Swelling	PWR Vessel Internals	Not Applicable
	Stainless Steel	Cracking/Stress Corrosion Cracking, Irradiation-Assisted Stress Corrosion Cracking	PWR Vessel Internals	Not Applicable
	Stainless Steel	Cracking/Stress Corrosion Cracking, Irradiation-Assisted Stress Corrosion Cracking	Water Chemistry	Not Applicable
	Stainless Steel	Loss of Material/Pitting and Crevice Corrosion	Water Chemistry	Not Applicable

Table B-1 Summary of SGS LRA Table 3.1.2-3 – Reactor Vessel Internals Summary of Aging Management Evaluation (cont.)

Component Type	Material	Aging Effect Requiring Management	Aging Management Programs	Comments
Upper Internals Assembly (head to vessel alignment pin bolt locking caps)	Stainless Steel	Changes in Dimensions/Void Swelling	PWR Vessel Internals	Note 2
	Stainless Steel	Cracking/Stress Corrosion Cracking, Irradiation-Assisted Stress Corrosion Cracking	PWR Vessel Internals	Note 2
	Stainless Steel	Cracking/Stress Corrosion Cracking, Irradiation-Assisted Stress Corrosion Cracking	Water Chemistry	Note 2
	Stainless Steel	Loss of Material/Pitting and Crevice Corrosion	Water Chemistry	Note 2
Upper Internals Assembly (head to vessel alignment pin bolts)	Stainless Steel Bolting	Changes in Dimensions/Void Swelling	PWR Vessel Internals	Not Applicable
	Stainless Steel Bolting	Cracking/Stress Corrosion Cracking, Irradiation-Assisted Stress Corrosion Cracking	PWR Vessel Internals	Not Applicable
	Stainless Steel Bolting	Cracking/Stress Corrosion Cracking, Irradiation-Assisted Stress Corrosion Cracking	Water Chemistry	Not Applicable
	Stainless Steel Bolting	Loss of Material/Pitting and Crevice Corrosion	Water Chemistry	Not Applicable
	Stainless Steel Bolting	Loss of Preload/Stress Relaxation	PWR Vessel Internals	Not Applicable

Table B-1 Summary of SGS LRA Table 3.1.2-3 – Reactor Vessel Internals Summary of Aging Management Evaluation (cont.)

Component Type	Material	Aging Effect Requiring Management	Aging Management Programs	Comments
Upper Internals Assembly (head to vessel alignment pins)	Stainless Steel	Changes in Dimensions/Void Swelling	PWR Vessel Internals	Not Applicable
	Stainless Steel	Cracking/Stress Corrosion Cracking, Irradiation-Assisted Stress Corrosion Cracking	PWR Vessel Internals	Not Applicable
	Stainless Steel	Cracking/Stress Corrosion Cracking, Irradiation-Assisted Stress Corrosion Cracking	Water Chemistry	Not Applicable
	Stainless Steel	Loss of Material/Pitting and Crevice Corrosion	Water Chemistry	Not Applicable
	Stainless Steel	Loss of Material/Wear	ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD	Not Applicable
Upper Internals Assembly (hold down spring)	Stainless Steel	Changes in Dimensions/Void Swelling	PWR Vessel Internals	Not Applicable
	Stainless Steel	Cracking/Stress Corrosion Cracking, Irradiation-Assisted Stress Corrosion Cracking	PWR Vessel Internals	Not Applicable
	Stainless Steel	Cracking/Stress Corrosion Cracking, Irradiation-Assisted Stress Corrosion Cracking	Water Chemistry	Not Applicable
	Stainless Steel	Loss of Material/Pitting and Crevice Corrosion	Water Chemistry	Not Applicable
	Stainless Steel	Loss of Preload/Stress Relaxation	PWR Vessel Internals	Not Applicable

Table B-1 Summary of SGS LRA Table 3.1.2-3 – Reactor Vessel Internals Summary of Aging Management Evaluation

(cont.)

Component Type	Material	Aging Effect Requiring Management	Aging Management Programs	Comments
Upper Internals Assembly (nuts)	Stainless Steel	Changes in Dimensions/Void Swelling	PWR Vessel Internals	Not Applicable
	Stainless Steel	Cracking/Stress Corrosion Cracking, Irradiation-Assisted Stress Corrosion Cracking	PWR Vessel Internals	Not Applicable
	Stainless Steel	Cracking/Stress Corrosion Cracking, Irradiation-Assisted Stress Corrosion Cracking	Water Chemistry	Not Applicable
	Stainless Steel	Loss of Material/Pitting and Crevice Corrosion	Water Chemistry	Not Applicable
Upper Internals Assembly (orifice plates)	Stainless Steel	Changes in Dimensions/Void Swelling	PWR Vessel Internals	Note 5
	Stainless Steel	Cracking/Stress Corrosion Cracking, Irradiation-Assisted Stress Corrosion Cracking	PWR Vessel Internals	Note 5
	Stainless Steel	Cracking/Stress Corrosion Cracking, Irradiation-Assisted Stress Corrosion Cracking	Water Chemistry	Note 5
	Stainless Steel	Loss of Material/Pitting and Crevice Corrosion	Water Chemistry	Note 5
Upper Internals Assembly (static flow mixers)	Cast Austenitic Stainless Steel (CASS)	Changes in Dimensions/Void Swelling	PWR Vessel Internals	Note 5
	Cast Austenitic Stainless Steel (CASS)	Cracking/Stress Corrosion Cracking, Irradiation-Assisted Stress Corrosion Cracking	PWR Vessel Internals	Note 5
	Cast Austenitic Stainless Steel (CASS)	Cracking/Stress Corrosion Cracking, Irradiation-Assisted Stress Corrosion Cracking	Water Chemistry	Note 5
	Cast Austenitic Stainless Steel (CASS)	Loss of Material/Pitting and Crevice Corrosion	Water Chemistry	Note 5

Table B-1 Summary of SGS LRA Table 3.1.2-3 – Reactor Vessel Internals Summary of Aging Management Evaluation

(cont.)

Component Type	Material	Aging Effect Requiring Management	Aging Management Programs	Comments
Upper Internals Assembly (upper core plate alignment pins)	Stainless Steel	Changes in Dimensions/Void Swelling	PWR Vessel Internals	Not Applicable
	Stainless Steel	Cracking/Stress Corrosion Cracking, Irradiation-Assisted Stress Corrosion Cracking	PWR Vessel Internals	Not Applicable
	Stainless Steel	Cracking/Stress Corrosion Cracking, Irradiation-Assisted Stress Corrosion Cracking	Water Chemistry	Not Applicable
	Stainless Steel	Loss of Material/Pitting and Crevice Corrosion	Water Chemistry	Not Applicable
	Stainless Steel	Loss of Material/Wear	ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD	Not Applicable
Upper Internals Assembly (upper core plate, insert, spacer ring, upper support plate, and upper support ring or skirt)	Stainless Steel	Changes in Dimensions/Void Swelling	PWR Vessel Internals	Not Applicable
	Stainless Steel	Cracking/Stress Corrosion Cracking, Irradiation-Assisted Stress Corrosion Cracking	PWR Vessel Internals	Not Applicable
	Stainless Steel	Cracking/Stress Corrosion Cracking, Irradiation-Assisted Stress Corrosion Cracking	Water Chemistry	Not Applicable
	Stainless Steel	Loss of Material/Pitting and Crevice Corrosion	Water Chemistry	Not Applicable

Table B-1 Summary of SGS LRA Table 3.1.2-3 – Reactor Vessel Internals Summary of Aging Management Evaluation (cont.)

Component Type	Material	Aging Effect Requiring Management	Aging Management Programs	Comments
Upper Internals Assembly (upper support column bases)	Cast Austenitic Stainless Steel (CASS)	Changes in Dimensions/Void Swelling	PWR Vessel Internals	Not Applicable
	Cast Austenitic Stainless Steel (CASS)	Cracking/Stress Corrosion Cracking, Irradiation-Assisted Stress Corrosion Cracking	PWR Vessel Internals	Not Applicable
	Cast Austenitic Stainless Steel (CASS)	Cracking/Stress Corrosion Cracking, Irradiation-Assisted Stress Corrosion Cracking	Water Chemistry	Not Applicable
	Cast Austenitic Stainless Steel (CASS)	Loss of Material/Pitting and Crevice Corrosion	Water Chemistry	Not Applicable
Upper Internals Assembly (upper support column bodies)	Stainless Steel	Changes in Dimensions/Void Swelling	PWR Vessel Internals	Not Applicable
	Stainless Steel	Cracking/Stress Corrosion Cracking, Irradiation-Assisted Stress Corrosion Cracking	PWR Vessel Internals	Not Applicable
	Stainless Steel	Cracking/Stress Corrosion Cracking, Irradiation-Assisted Stress Corrosion Cracking	Water Chemistry	Not Applicable
	Stainless Steel	Loss of Material/Pitting and Crevice Corrosion	Water Chemistry	Not Applicable
Upper Internals Assembly (upper support column bolts)	Stainless Steel Bolting	Changes in Dimensions/Void Swelling	PWR Vessel Internals	Not Applicable
	Stainless Steel Bolting	Cracking/Stress Corrosion Cracking, Irradiation-Assisted Stress Corrosion Cracking	PWR Vessel Internals	Not Applicable
	Stainless Steel Bolting	Cracking/Stress Corrosion Cracking, Irradiation-Assisted Stress Corrosion Cracking	Water Chemistry	Not Applicable
	Stainless Steel Bolting	Loss of Material/Pitting and Crevice Corrosion	Water Chemistry	Not Applicable
	Stainless Steel Bolting	Loss of Preload/Stress Relaxation	PWR Vessel Internals	Not Applicable

Table B-1 Summary of SGS LRA Table 3.1.2-3 – Reactor Vessel Internals Summary of Aging Management Evaluation

(cont.)

Component Type	Material	Aging Effect Requiring Management	Aging Management Programs	Comments
Upper Internals Assembly (upper support column extension tubes and adapters)	Stainless Steel	Changes in Dimensions/Void Swelling	PWR Vessel Internals	Not Applicable
	Stainless Steel	Cracking/Stress Corrosion Cracking, Irradiation-Assisted Stress Corrosion Cracking	PWR Vessel Internals	Not Applicable
	Stainless Steel	Cracking/Stress Corrosion Cracking, Irradiation-Assisted Stress Corrosion Cracking	Water Chemistry	Not Applicable
	Stainless Steel	Loss of Material/Pitting and Crevice Corrosion	Water Chemistry	Not Applicable
Upper Internals Assembly (upper support column flanges)	Stainless Steel	Changes in Dimensions/Void Swelling	PWR Vessel Internals	Not Applicable
	Stainless Steel	Cracking/Stress Corrosion Cracking, Irradiation-Assisted Stress Corrosion Cracking	PWR Vessel Internals	Not Applicable
	Stainless Steel	Cracking/Stress Corrosion Cracking, Irradiation-Assisted Stress Corrosion Cracking	Water Chemistry	Not Applicable
	Stainless Steel	Loss of Material/Pitting and Crevice Corrosion	Water Chemistry	Not Applicable

Table B-1 Summary of SGS LRA Table 3.1.2-3 – Reactor Vessel Internals Summary of Aging Management Evaluation (cont.)

Component Type	Material	Aging Effect Requiring Management	Aging Management Programs	Comments
Upper Internals Assembly (upper support column lock keys)	Stainless Steel	Changes in Dimensions/Void Swelling	PWR Vessel Internals	Not Applicable
	Stainless Steel	Cracking/Stress Corrosion Cracking, Irradiation-Assisted Stress Corrosion Cracking	PWR Vessel Internals	Not Applicable
	Stainless Steel	Cracking/Stress Corrosion Cracking, Irradiation-Assisted Stress Corrosion Cracking	Water Chemistry	Not Applicable
	Stainless Steel	Loss of Material/Pitting and Crevice Corrosion	Water Chemistry	Not Applicable
	Stainless Steel	Loss of Material/Wear	ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD	Not Applicable

Note:

1. During the evaluations performed to address A/LAI 1, 2, and 7, it was identified that this component for SGS Unit 1 is fabricated from CASS material. See Section 6.2.7 and Table 6-2 for additional information.
2. These components and associated aging effects requiring management are not applicable to SGS Unit 1. These components were listed within Table 3.1.2-3 of the SGS LRA but do not exist within SGS Unit 1 [41]. These components have been left in this table to maintain consistency with [1].
3. During the evaluations performed to address A/LAI 1 and 2, it was identified that this component for SGS Unit 1 is fabricated from 304 stainless steel. Therefore, this component was not evaluated to address A/LAI 7 in Section 6.2.7.
4. During the evaluations performed to address A/LAI 1 and 2, it was identified that one of the components in this category for SGS Unit 1, specifically the "Lower Support Column – Cap," is fabricated from CASS material See Section 6.2.7 and Table 6-2 for additional information. No components within this category were identified to be fabricated from nickel alloy.
5. During the evaluations performed to address A/LAI 1, 2, and 7, it was identified that this component for SGS Unit 1 is fabricated from CASS material and 304 stainless steel. See Section 6.2.7 and Table 6-2 for additional information.
6. During the evaluations performed to address A/LAI 1 and 2, it was identified that this component for SGS Unit 1 is fabricated from Alloy 600.
7. This component type is the same as the "Lower Internal Assembly (lower radial support keys)" as identified in the SGS Unit 1 AMR (Lower Internals Assembly – Radial Support Keys per MRP-191 [14]).

APPENDIX C
MRP-227-A AUGMENTED INSPECTIONS

Table C-1 MRP-227-A Primary Inspection and Monitoring Recommendations for Westinghouse-Designed Internals

Item	Applicability	Effect (Mechanism)	Expansion Link (Note 1)	Examination Method/Frequency (Note 1)	Examination Coverage
Control Rod Guide Tube Assembly Guide plates (cards)	All plants	Loss of Material (wear)	None	Visual (VT-3) Per the schedule requirements of WCAP-17451-P Section 5 including subsequent examinations. (Note 7)	Minimum examination of 20% of the number of CRGT assemblies, and as per the requirements of WCAP-17451-P Revision 1 Section 5. (Note 7) See Figure A-2
Control Rod Guide Tube Assembly Lower flange welds	All plants	Cracking (SCC, Fatigue) Aging Management (IE and TE)	Bottom-mounted instrumentation (BMI) column bodies, lower support column bodies (cast), Upper core plate, Lower support forging/casting	Enhanced visual (EVT-1) examination to determine the presence of crack-like surface flaws in flange welds no later than 2 refueling outages from the beginning of the license renewal period and subsequent examination on a ten-year interval.	100% of outer (accessible) CRGT lower flange weld surfaces and adjacent base metal on the individual periphery CRGT assemblies. (Note 2) See Figure A-3.
Core Barrel Assembly Upper core barrel flange weld	All plants	Cracking (SCC)	Lower support column bodies (non-cast) Core barrel outlet nozzle welds	Periodic enhanced visual (EVT-1) examination, no later than 2 refueling outages from the beginning of the license renewal period and subsequent examination on a ten-year interval.	100% of one side of the accessible surfaces of the selected weld and adjacent base metal (Note 4). See Figure A-4.

Table C-1 (cont.) MRP-227-A Primary Inspection and Monitoring Recommendations for Westinghouse-Designed Internals

Item	Applicability	Effect (Mechanism)	Expansion Link (Note 1)	Examination Method/Frequency (Note 1)	Examination Coverage
Core Barrel Assembly Upper and lower core barrel cylinder girth welds	All plants	Cracking (SCC, IASCC, Fatigue)	Upper and lower core barrel cylinder axial welds	Periodic enhanced visual (EVT-1) examination, no later than 2 refueling outages from the beginning of the license renewal period and subsequent examinations on a ten-year interval.	100% of one side of the accessible surfaces of the selected weld and adjacent base metal (Note 4). See Figure A-4
Core Barrel Assembly Lower core barrel flange weld (Note 5)	All plants	Cracking (SCC, Fatigue)	None	Periodic enhanced visual (EVT-1) examination, no later than 2 refueling outages from the beginning of the license renewal period and subsequent examinations on a ten-year interval.	100% of one side of the accessible surfaces of the selected weld and adjacent base metal (Note 4).
Baffle-Former Assembly Baffle-edge bolts	All plants with baffle-edge bolts NOTE: Applicable to SGS Unit 1	Cracking (IASCC, Fatigue) that results in • Lost or broken locking devices • Failed or missing bolts • Protrusion of bolt heads Aging Management (IE and ISR) (Note 6)	None	Visual (VT-3) examination, with baseline examination between 20 and 40 EFPY and subsequent examinations on a ten-year interval.	Bolts and locking devices on high fluence seams. 100% of components accessible from core side (Note 3). See Figures A-5 and A-7.

Table C-1 MRP-227-A Primary Inspection and Monitoring Recommendations for Westinghouse-Designed Internals (cont.)

Item	Applicability	Effect (Mechanism)	Expansion Link (Note 1)	Examination Method/Frequency (Note 1)	Examination Coverage
Baffle-Former Assembly Baffle-former bolts	All plants	Cracking (IASCC, Fatigue) Aging Management (IE and ISR) (Note 6)	Lower support column bolts, Barrel-former bolts	Baseline volumetric (UT) examination between 25 and 35 EFPY, with subsequent examinations on a ten-year interval. (Note 8)	100% of accessible bolts (Note 3). Heads accessible from the core side. UT accessibility may be affected by complexity of head and locking device designs. See Figures A-5 and A-7.

Table C-1 (cont.) MRP-227-A Primary Inspection and Monitoring Recommendations for Westinghouse-Designed Internals

Item	Applicability	Effect (Mechanism)	Expansion Link (Note 1)	Examination Method/Frequency (Note 1)	Examination Coverage
Baffle-Former Assembly Assembly (Includes: Baffle plates, baffle edge bolts and indirect effects of void swelling in former plates)	All plants	Distortion (Void Swelling), or Cracking (IASCC) that results in: • Abnormal interaction with fuel assemblies • Gaps along high fluence baffle joint • Vertical displacement of baffle plates near high fluence joint • Broken or damaged edge bolt locking systems along high fluence baffle joint	None	Visual (VT-3) examination to check for evidence of distortion, with baseline examination between 20 and 40 EFPY and subsequent examinations on a ten-year interval.	Core side surface as indicated. See Figures A-5 and A-8.

Table C-1 (cont.) MRP-227-A Primary Inspection and Monitoring Recommendations for Westinghouse-Designed Internals

Item	Applicability	Effect (Mechanism)	Expansion Link (Note 1)	Examination Method/Frequency (Note 1)	Examination Coverage
Alignment and Interfacing Components Internals hold down spring	All plants with 304 stainless steel hold down springs NOTE: SGS Unit 1 hold down spring is 304 SS	Distortion (Loss of Load) Note: This mechanism was not strictly identified in the original list of age-related degradation mechanisms.	None	Direct measurement of spring height within three cycles of the beginning of the license renewal period. If the first set of measurements is not sufficient to determine life, spring height measurements must be taken during the next two outages, in order to extrapolate the expected spring height to 60 years.	Measurements should be taken at several points around the circumference of the spring, with a statistically adequate number of measurements at each point to minimize uncertainty. See Figure A-9.
Thermal Shield Assembly Thermal shield flexures	All plants with thermal shields NOTE: Applicable to SGS Unit 1	Cracking (Fatigue) or Loss of Material (Wear) that results in thermal shield flexures excessive wear, fracture, or complete separation	None	Visual (VT-3) no later than 2 refueling outages from the beginning of the license renewal period. Subsequent examinations on a ten-year interval.	100% of thermal shield flexures. See Figures A-4 and A-10.

Table C-1 (cont.) MRP-227-A Primary Inspection and Monitoring Recommendations for Westinghouse-Designed Internals

Item	Applicability	Effect (Mechanism)	Expansion Link (Note 1)	Examination Method/Frequency (Note 1)	Examination Coverage

Notes:

1. Examination acceptance criteria and expansion criteria for the Westinghouse components are in Table C-4.
2. A minimum of 75% of the total identified sample population must be examined.
3. A minimum of 75% of the total population (examined + unexamined), including coverage consistent with the Expansion criteria in C-4, must be examined for inspection credit.
4. A minimum of 75% of the total weld length (examined + unexamined), including coverage consistent with the Expansion criteria in C-4, must be examined from either the inner or outer diameter for inspection credit.
5. The lower core barrel flange weld may be alternatively designated as the core barrel-to-support plate weld in some Westinghouse plant designs.
6. Void swelling effects on this component is managed through management of void swelling on the entire baffle-former assembly.
7. WCAP-17451-P Revision 1 [44] requires a remote visual examination consistent with visual (VT-3) for minimum compliance and examination coverage of a minimum of 20% of the number of CRGT guide card assemblies. The baseline examination schedule has been adjusted for various CRGT designs, the extent of individual CRGT examination modified, and flexible subsequent examination regimens correlating to initial baseline sample size, accuracy of wear estimation and examination results. The content of this note and the noted cells is included according to the interim guidance provided in the EPRI transmittal MRP 2014-006 [48].
8. Baffle-former bolt inspection frequency is 10 years following the initial or baseline inspection, unless SGS Unit 1 provides an evaluation for NRC staff approval that justifies a longer interval between inspections.

Table C-2 MRP-227-A Expansion Inspection and Monitoring Recommendations for Westinghouse-Designed Internals

Item	Applicability	Effect (Mechanism)	Primary Link (Note 1)	Examination Method/Frequency (Note 1, Note 3)	Examination Coverage
Upper Internals Assembly Upper Core Plate	All plants	Cracking (Fatigue, Wear)	CRGT lower flange weld	Enhanced visual (EVT-1) examination. Re-inspection every 10 years following initial inspection.	100% of accessible surfaces (Note 2).
Lower Internals Assembly Lower support forging or castings	All plants NOTE: SGS Unit 1 has a lower support casting	Cracking Aging Management (TE in Casting)	CRGT lower flange weld	Enhanced visual (EVT-1) examination. Re-inspection every 10 years following initial inspection.	100% of accessible surfaces (Note 2). See Figure A-12.
Core Barrel Assembly Barrel-former bolts	All plants	Cracking (IASCC, Fatigue) Aging Management (IE, Void Swelling and ISR)	Baffle-former bolts	Volumetric (UT) examination. Re-inspection every 10 years following initial inspection.	100% of accessible bolts. Accessibility may be limited by presence of thermal shields or neutron pads (Note 2). See Figure A-7.
Lower Support Assembly Lower support column bolts	All plants	Cracking (IASCC, Fatigue) Aging Management (IE and ISR)	Baffle-former bolts	Volumetric (UT) examination. Re-inspection every 10 years following initial inspection.	100% of accessible bolts or as supported by plant-specific justification (Note 2). See Figures A-11and A-12.

Table C-2 (cont.) MRP-227-A Expansion Inspection and Monitoring Recommendations for Westinghouse-Designed Internals

Item	Applicability	Effect (Mechanism)	Primary Link (Note 1)	Examination Method/Frequency (Note 1, Note 3)	Examination Coverage
Core Barrel Assembly Core barrel outlet nozzle welds	All plants	Cracking (SCC, Fatigue) Aging Management (IE of lower sections)	Upper core barrel flange weld	Enhanced visual (EVT-1) examination. Re-inspection every 10 years following initial inspection.	100% of one side of the accessible surfaces of the selected weld and adjacent base metal (Note 2). See Figure A-4.
Core Barrel Assembly Upper and lower core barrel cylinder axial welds	All plants	Cracking (SCC, IASCC) Aging Management (IE)	Upper and lower core barrel cylinder girth welds	Enhanced visual (EVT-1) examination. Re-inspection every 10 years following initial inspection.	100% of one side of the accessible surfaces of the selected weld and adjacent base metal (Note 2). See Figure A-4.
Lower Support Assembly Lower support column bodies (non cast)	All plants	Cracking (IASCC) Aging Management (IE)	Upper core barrel flange weld	Enhanced visual (EVT-1) examination. Re-inspection every 10 years following initial inspection.	100% of accessible surfaces (Note 2). See Figures A-12 and A-13.
Lower Support Assembly Lower support column bodies (cast)	All plants	Cracking (IASCC) including the detection of fractured support columns Aging Management (IE)	Control rod guide tube (CRGT) lower flanges	Visual (EVT-1) examination. Re-inspection every 10 years following initial inspection.	100% of accessible support columns (Note 2). See Figures A-12 and A-13.

Table C-2 (cont.) MRP-227-A Expansion Inspection and Monitoring Recommendations for Westinghouse-Designed Internals

Item	Applicability	Effect (Mechanism)	Primary Link (Note 1)	Examination Method/Frequency (Note 1, Note 3)	Examination Coverage
Bottom-Mounted Instrumentation System Bottom-mounted instrumentation (BMI) column bodies	All plants	Cracking (Fatigue) including the detection of completely fractured column bodies Aging Management (IE)	Control rod guide tube (CRGT) lower flanges	Visual (VT-3) examination of BMI column bodies as indicated by difficulty of insertion/withdrawal of flux thimbles. Re-inspection every 10 years following initial inspection. Flux thimble insertion/withdrawal to be monitored at each inspection interval.	100% of BMI column bodies for which difficulty is detected during flux thimble insertion/withdrawal. See Figures A-12 and A-14.

Notes:

1. Examination acceptance criteria and expansion criteria for the Westinghouse components are in Table C-4.
2. A minimum of 75% coverage of the entire examination area or volume, or a minimum sample size of 75% of the total population of like components of the examination is required (including both the accessible and inaccessible portions).
3. Expansion component inspection frequency is 10 years following the initial or baseline inspection, unless SGS Unit 1 provides an evaluation for NRC staff approval that justifies a longer interval between inspections.

Table C-3 MRP-227-A Existing Inspection and Aging Management Programs Credited in Recommendations for Westinghouse-Designed Internals

Item	Applicability	Effect (Mechanism)	Primary Link	Examination Method	Examination Coverage
Core Barrel Assembly Core barrel flange	All plants	Loss of material (Wear)	ASME Code Section XI	Visual (VT-3) examination to determine general condition for excessive wear.	All accessible surfaces at specified frequency.
Upper Internals Assembly Upper support ring or skirt	All plants	Cracking (SCC, Fatigue)	ASME Code Section XI	Visual (VT-3) examination.	All accessible surfaces at specified frequency.
Lower Internals Assembly Lower core plate XL lower core plate (Note 1)	All plants	Cracking (IASCC, Fatigue) Aging Management (IE)	ASME Code Section XI	Visual (VT-3) examination of the lower core plates to detect evidence of distortion and/or loss of bolt integrity.	All accessible surfaces at specified frequency.
Lower Internals Assembly Lower core plate XL lower core plate (Note 1)	All plants	Loss of material (Wear)	ASME Code Section XI	Visual (VT-3) examination.	All accessible surfaces at specified frequency.
Bottom-Mounted Instrumentation System Flux thimble tubes	All plants	Loss of material (Wear)	NUREG-1801, Rev. 1	Surface (ET) examination.	Eddy current surface examination, as defined in plant response to IEB 88-09.
Alignment and Interfacing Components Clevis insert bolts	All plants	Loss of material (Wear) (Note 2)	ASME Code Section XI	Visual (VT-3) examination.	All accessible surfaces at specified frequency.

Table C-3 (cont.) MRP-227-A Existing Inspection and Aging Management Programs Credited in Recommendations for Westinghouse-Designed Internals

Item	Applicability	Effect (Mechanism)	Primary Link	Examination Method	Examination Coverage
Alignment and Interfacing Components Upper core plate alignment pins	All plants	Loss of material (Wear)	ASME Code Section XI	Visual (VT-3) examination.	All accessible surfaces at specified frequency.

Notes:
1. XL = "Extra Long," referring to Westinghouse plants with 14-foot cores.
2. Bolt was screened in because of stress relaxation and associated cracking; however, wear of the clevis/insert is the issue.

Table C-4 MRP-227-A Acceptance Criteria and Expansion Criteria Recommendations for Westinghouse-Designed Internals

Item	Applicability	Examination Acceptance Criteria (Note 1)	Expansion Link(s)	Expansion Criteria	Additional Examination Acceptance Criteria
Control Rod Guide Tube Assembly Guide plates (cards)	All plants	Visual (VT-3) examination. (Note 3) The specific relevant condition is wear that could lead to loss of control rod alignment and impede control assembly insertion.	None	N/A	N/A

Table C-4 (cont.) MRP-227-A Acceptance Criteria and Expansion Criteria Recommendations for Westinghouse-Designed Internals

Item	Applicability	Examination Acceptance Criteria (Note 1)	Expansion Link(s)	Expansion Criteria	Additional Examination Acceptance Criteria
Control Rod Guide Tube Assembly Lower flange welds	All plants	Enhanced visual (EVT-1) examination. The specific relevant condition is a detectable crack-like surface indication.	a. Bottom-mounted instrumentation (BMI) column bodies b. Lower support column bodies (cast), upper core plate and lower support forging or casting	a. Confirmation of surface-breaking indications in two or more CRGT lower flange welds, combined with flux thimble insertion/withdrawal difficulty, shall require visual (VT-3) examination of BMI column bodies by the completion of the next refueling outage. b. Confirmation of surface-breaking indications in two or more CRGT lower flange welds shall require EVT-1 examination of cast lower support column bodies, upper core plate and lower support forging/castings within three fuel cycles following the initial observation.	a. For BMI column bodies, the specific relevant condition for the VT-3 examination is completely fractured column bodies. b. For cast lower support column bodies, upper core plate and lower support forging/castings, the specific relevant condition is a detectable crack-like surface indication.

Table C-4 (cont.) MRP-227-A Acceptance Criteria and Expansion Criteria Recommendations for Westinghouse-Designed Internals

Item	Applicability	Examination Acceptance Criteria (Note 1)	Expansion Link(s)	Expansion Criteria	Additional Examination Acceptance Criteria
Core Barrel Assembly Upper core barrel flange weld	All plants	Periodic enhanced visual (EVT-1) examination. The specific relevant condition is a detectable crack-like surface indication.	a. Core barrel outlet nozzle welds b. Lower support column bodies (non cast)	a. The confirmed detection and sizing of a surface-breaking indication with a length greater than two inches in the upper core barrel flange weld shall require that the EVT-1 examination be expanded to include the core outlet nozzle welds by the completion of the next refueling outage. b. If extensive cracking in the core barrel outlet nozzle welds is detected, EVT-1 examination shall be expanded to include the upper six inches of the accessible surfaces of the non-cast lower support column bodies within three fuel cycles follow the initial observation.	a and b. The specific relevant condition for the expansion core barrel outlet nozzle weld and lower support column body examination is a detectable crack-like surface indication.

Table C-4 (cont.) MRP-227-A Acceptance Criteria and Expansion Criteria Recommendations for Westinghouse-Designed Internals

Item	Applicability	Examination Acceptance Criteria (Note 1)	Expansion Link(s)	Expansion Criteria	Additional Examination Acceptance Criteria
Core Barrel Assembly Lower core barrel flange weld (Note 2)	All plants	Periodic enhanced visual (EVT-1) examination. The specific relevant condition is a detectable crack-like surface indication.	None	None	None
Core Barrel Assembly Upper core barrel cylinder girth welds	All plants	Periodic enhanced visual (EVT-1) examination. The specific relevant condition is a detectable crack-like surface indication.	Upper core barrel cylinder axial welds	The confirmed detection and sizing of a surface-breaking indication with a length greater than two inches in the upper core barrel cylinder girth welds shall require that the EVT-1 examination be expanded to include the upper core barrel cylinder axial welds by the completion of the next refueling outage.	The specific relevant condition for the expansion upper core barrel cylinder axial weld examination is a detectable crack-like surface indication.

Table C-4 (cont.) MRP-227-A Acceptance Criteria and Expansion Criteria Recommendations for Westinghouse-Designed Internals

Item	Applicability	Examination Acceptance Criteria (Note 1)	Expansion Link(s)	Expansion Criteria	Additional Examination Acceptance Criteria
Core Barrel Assembly Lower core barrel cylinder girth welds	All plants	Periodic enhanced visual (EVT-1) examination. The specific relevant condition is a detectable crack-like surface indication.	Lower core barrel cylinder axial welds	The confirmed detection and sizing of a surface-breaking indication with a length greater than two inches in the lower core barrel cylinder girth welds shall require that the EVT-1 examination be expanded to include the lower core barrel cylinder axial welds by the completion of the next refueling outage.	The specific relevant condition for the expansion lower core barrel cylinder axial weld examination is a detectable crack-like surface indication.
Baffle-Former Assembly Baffle-edge bolts	All plants with baffle-edge bolts NOTE: Applicable to SGS Unit 1	Visual (VT-3) examination. The specific relevant conditions are missing or broken locking devices, failed or missing bolts, and protrusion of bolt heads.	None	N/A	N/A

Table C-4 (cont.) MRP-227-A Acceptance Criteria and Expansion Criteria Recommendations for Westinghouse-Designed Internals

Item	Applicability	Examination Acceptance Criteria (Note 1)	Expansion Link(s)	Expansion Criteria	Additional Examination Acceptance Criteria
Baffle-Former Assembly Baffle-former bolts	All plants	Volumetric (UT) examination. The examination acceptance criteria for the UT of the baffle-former bolts shall be established as part of the examination technical justification.	a. Lower support column bolts b. Barrel-former bolts	a. Confirmation that more than 5% of the baffle-former bolts actually examined on the four baffle plates at the largest distance from the core (presumed to be the lowest dose locations) contain unacceptable indications shall require UT examination of the lower support column bolts within the next three fuel cycles. b. Confirmation that more than 5% of the lower support column bolts actually examined contain unacceptable indications shall require UT examination of the barrel-former bolts.	a and b. The examination acceptance criteria for the UT of the lower support column bolts and the barrel-former bolts shall be established as part of the examination technical justification.

Table C-4 (cont.) MRP-227-A Acceptance Criteria and Expansion Criteria Recommendations for Westinghouse-Designed Internals

Item	Applicability	Examination Acceptance Criteria (Note 1)	Expansion Link(s)	Expansion Criteria	Additional Examination Acceptance Criteria
Baffle-Former Assembly Assembly	All plants	Visual (VT-3) examination. The specific relevant conditions are evidence of abnormal interaction with fuel assemblies, gaps along high fluence shroud plate joints, vertical displacement of shroud plates near high fluence joints, and broken or damaged edge bolt locking systems along high fluence baffle plate joints.	None	N/A	N/A

Table C-4 (cont.) MRP-227-A Acceptance Criteria and Expansion Criteria Recommendations for Westinghouse-Designed Internals

Item	Applicability	Examination Acceptance Criteria (Note 1)	Expansion Link(s)	Expansion Criteria	Additional Examination Acceptance Criteria
Alignment and Interfacing Components Internals hold down spring	All plants with 304 stainless steel hold down springs NOTE: SGS Unit 1 hold down spring is 304 SS	Direct physical measurement or spring height. The examination acceptance criterion for this measurement is that the remaining compressible height of the spring shall provide hold-down forces within the plant-specific design tolerance.	None	N/A	N/A

Table C-4 MRP-227-A Acceptance Criteria and Expansion Criteria Recommendations for Westinghouse-Designed Internals (cont.)

Item	Applicability	Examination Acceptance Criteria (Note 1)	Expansion Link(s)	Expansion Criteria	Additional Examination Acceptance Criteria
Thermal Shield Assembly Thermal shield flexures	All plants with thermal shields NOTE: Applicable to SGS Unit 1	Visual (VT-3) examination. The specific relevant conditions for thermal shield flexures are excessive wear, fracture, or complete separation.	None	N/A	N/A

Note:

1. The examination acceptance criterion for visual examination is the absence of the specified relevance condition(s).
2. The lower core barrel flange weld may alternatively be designated as the core barrel-to-support plate weld in some Westinghouse plant designs.
3. WCAP-17451-P Revision 1 [44] specifies a remote visual examination consistent with visual (VT-3) but allows for various supplemental measurement techniques which if employed increase wear estimate accuracy and allow use of acceptance criteria (wear projections) to determine the appropriate re-examination interval. The content of this note and the noted cell is included according to the interim guidance provided in the EPRI transmittal MRP 2014-006 [48].

Attachment 2

WCAP-17438-NP, Revision 2, “PWR Vessel Internals Program Plan for Aging Management of Reactor Internals at Salem Generating Station, Unit 2”

Westinghouse Non-Proprietary Class 3

WCAP-17438-NP
Revision 2

July 2014

PWR Vessel Internals Program Plan for Aging Management of Reactor Internals at Salem Nuclear Generating Station, Salem Unit 2

Westinghouse

WCAP-17438-NP
Revision 2

PWR Vessel Internals Program Plan for Aging Management of Reactor Internals at Salem Nuclear Generating Station, Salem Unit 2

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LIST OF ACRONYMS

A/LAI	Applicant/Licensee Action Item
AMP	Aging Management Program Plan
AMR	Aging Management Review
ARDM	age-related degradation mechanism
ASME	American Society of Mechanical Engineers
B&PV	Boiler and Pressure Vessel
B&W	Babcock & Wilcox
BMI	bottom-mounted instrumentation
BWR	boiling water reactor
CASS	cast austenitic stainless steel
CE	Combustion Engineering
CFR	Code of Federal Regulations
CLB	current licensing basis
CMTR	certified material test report
CPE	corporate program engineer
CRGT	control rod guide tube
ECT	eddy current testing
EFPY	effective full-power years
EPRI	Electric Power Research Institute
ET	electromagnetic testing (eddy current)
EVT	enhanced visual testing (a visual NDE method that includes EVT-1)
FMECA	failure modes, effects, and criticality analysis
GALL	Generic Aging Lessons Learned
I&E	Inspection and Evaluation
IASCC	irradiation-assisted stress corrosion cracking
IE	irradiation embrittlement
INPO	Institute of Nuclear Power Operations
ISI	inservice inspection
ISR	irradiation-enhanced stress relaxation
LRA	license renewal application
LRAAI	license renewal applicant action items
MRP	Materials Reliability Program
NDE	nondestructive examination
NEI	Nuclear Energy Institute
NOS	Nuclear Oversight Section
NRC	Nuclear Regulatory Commission
NSSS	nuclear steam supply system
OE	Operating Experience
OEM	Original Equipment Manufacturer
OER	Operating Experience Report
PBD	program basis document
PH	precipitation-hardening
PSEG	Public Service Enterprise Group
PWR	pressurized water reactor

LIST OF ACRONYMS (cont.)

PWROG	Pressurized Water Reactor Owners Group
PWSCC	primary water stress corrosion cracking
QA	quality assurance
RCS	reactor coolant system
RIS	Regulatory Issue Summary
RO	refueling outage
RVI	reactor vessel internals
SCC	stress corrosion cracking
SE	Safety Evaluation
SER	Safety Evaluation Report
SGS	Salem Nuclear Generating Station
SPE	site program engineer
SRP	Standard Review Plan
SS	stainless steel
SSC	systems, structures, and components
TE	thermal embrittlement
TLAA	time-limited aging analysis
UFSAR	Updated Final Safety Analysis Report
U.S.	United States
UT	ultrasonic testing (a volumetric NDE method)
VT	visual testing (a visual NDE method that includes VT-1 and VT-3)
WOG	Westinghouse Owners Group
XL	extra-long Westinghouse fuel

Trademark Statements:

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1 PURPOSE

The purpose of this report is to document the Salem Nuclear Generating Station (SGS), Unit 2, hereafter SGS Unit 2, Reactor Internals Aging Management Program Plan (AMP). This revision updates the report to incorporate the latest revision of Reference [46].

Public Services Enterprise Group (PSEG) Nuclear LLC is a Delaware limited liability company formed to own and operate nuclear generating stations. PSEG Nuclear LLC is a wholly owned subsidiary of PSEG Power LLC, which is wholly owned by PSEG Incorporated, a corporation formed under the laws of the State of New Jersey. PSEG Nuclear LLC is the licensed operator of SGS Unit 2.

The purpose of the AMP is to manage the effects of aging on the reactor internals of SGS Unit 2 through the period of extended operation. SGS Unit 2 enters the period of extended operation at midnight on April 18, 2020 [1]. This document provides a description of the program as it relates to the management of aging effects identified in various regulatory and updated industry-generated documents, in addition to the program documented in SGS Unit 2 procedure document ER-AP-333 [2], for reactor internals. It is prepared in accordance with the various regulatory and industry-generated documents, and is supported by existing SGS documents and procedures. As required by industry experience or directive in the future, it will be updated or supported by additional documents to provide clear and concise direction for the effective management of aging degradation in reactor internals components. These actions provide assurance that operations at SGS Unit 2 will continue to be conducted in accordance with the current licensing basis (CLB) for the reactor internals by fulfilling license renewal commitments [3], United States (U.S.) Nuclear Regulatory Commission (NRC) expectations in the Regulatory Issue Summary (RIS) [4], following American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel (B&PV) Code Section XI Inservice Inspection (ISI) requirements [5], and meeting industry requirements [6]. This AMP fully captures the intent of the additional industry guidance for reactor internals augmented inspections, based on the programs sponsored by U.S. utilities through the Electric Power Research Institute (EPRI) managed Materials Reliability Program (MRP) and the Pressurized Water Reactor Owners Group (PWROG).

The main objectives for the SGS Unit 2 reactor vessel internals (RVI) AMP are to:

- Demonstrate that the effects of aging on the reactor internals will be adequately managed for the period of extended operation in accordance with 10 CFR 54 [7].
- Summarize the role of existing SGS Unit 2 aging management programs in the reactor vessel internals AMP.
- Define and implement the industry-defined (EPRI/MRP and PWROG) pressurized water reactor (PWR) reactor vessel internals requirements and guidance for managing aging effects on reactor internals.
- Provide an inspection plan summary for the SGS Unit 2 reactor internals.

The Safety Evaluation Report (SER) for SGS Unit 2 license renewal [3] includes the license renewal commitment for the reactor vessel internals. This commitment is also documented in the SGS aging

management program basis document (PBD) [8] and SGS Updated Final Safety Analysis Report (UFSAR) Appendix B, Section A.2.1.7, "PWR Vessel Internals" [9].

SGS License Renewal Commitment 7 addresses the creation of a program for the PWR Reactor Vessel Internals at SGS Unit 2.

Commitment 7 [3]: PWR Vessel Internals is a new program that will include the following activities:

1. Participate in the industry programs for investigating and managing aging effects on reactor internals.
2. Evaluate and implement the results of the industry programs as applicable to the reactor internals.
3. Upon completion of these programs, but not less than 24 months before entering the period of extended operation, submit an inspection plan for reactor internals to the NRC for review and approval.

The SER also included commitments related to reactor vessel internals. The purpose of these commitments is to continue the ongoing programs for ASME XI ISIs, Subsections IWB, IWC, and IWD (License Renewal Commitment 1) and water chemistry (License Renewal Commitment 2) and to implement a new program for flux thimble tube inspection (License Renewal Commitment 25).

Augmented inspections, based on required program enhancements resulting from industry programs [6], will become part of the SGS Unit 2 ISI program [10]. Corrective actions for augmented inspections will be developed and will use repair and replacement procedures equivalent to those requirements in ASME B&PV Code, Section XI [5], or they will use processes determined to be equivalent to or more rigorous than currently defined procedures as determined independently by PSEG Nuclear, or in cooperation with the industry.

This AMP for the SGS Unit 2 reactor internals demonstrates that the program adequately manages the effects of aging for reactor internals components. The AMP establishes the basis for providing reasonable assurance that the internals components will continue to perform their intended function through the SGS Unit 2 period of extended operation. This AMP supports the SGS Unit 2 License Renewal Commitment 7 to submit a PWR Vessel Internals inspection plan to the NRC, as it will be implemented from PSEG Nuclear participation in industry initiatives, 24 months prior to the period of extended operation. Thus, the program must be submitted no later than April 18, 2018.

The development and implementation of this program meets the guidelines provided in the RIS [4] by supporting the commitment to submit a PWR Vessel Internals inspection plan in accordance with MRP-227-A for SGS Unit 2.

2 BACKGROUND

The management of aging degradation effects in reactor internals is required for nuclear plants considering or entering license renewal, as specified in the NRC Standard Review Plan (SRP) [11]. The U.S. nuclear power industry has been actively engaged in recent years in a significant effort to support the industry goal of responding to these requirements. Various programs have been underway within the industry over the past decade to develop guidelines for managing the effects of aging within PWR reactor internals. In 1997, the Westinghouse Owners Group (formerly WOG, now PWROG) issued WCAP-14577 [12], "License Renewal Evaluation: Aging Management for Reactor Internals," which was reissued as Revision 1-A in 2001 after receiving NRC staff review and approval. Later, an effort was engaged by the EPRI MRP to address the PWR internals aging management issue for the three currently operating U.S. reactor designs - Westinghouse, Combustion Engineering (CE), and Babcock & Wilcox (B&W).

The MRP first established a framework and strategy for the aging management of PWR internals components using proven and familiar methods for inspection, monitoring, surveillance, and communication. Based upon that framework and strategy, as well as accumulated industry research data, the following elements of an AMP were further developed [12, 14]:

- Screening criteria were developed, considering chemical composition, neutron fluence exposure, temperature history, and representative stress levels, for determining the relative susceptibility of PWR internals components to each of eight postulated aging mechanisms (further discussed in Section 4 of this program).

- PWR internals components were categorized, based on the screening criteria, into categories that ranged from:

  - Components for which the effects from the postulated aging mechanisms are insignificant,
  - Components that are moderately susceptible to the aging effects, and
  - Components that are significantly susceptible to the aging effects.

- Functionality assessments were performed to determine the effects of the degradation mechanisms on component functionality. These assessments were based on representative plant designs of PWR internals components and assemblies of components using irradiated and aged material properties.

Aging management strategies were developed by combining the results of the functionality assessment with several contributing factors to determine the appropriate aging management methodology, baseline examination timing, and the need and timing of subsequent inspections. Items considered included component accessibility, operating experience (OE), existing evaluations, and prior examination results.

The industry guidance is contained within two separate EPRI MRP documents:

- MRP-227-A [6], "Pressurized Water Reactor Internals Inspection and Evaluation Guidelines," (hereafter referred to as "the Inspection and Evaluation (I&E) Guidelines" or simply "MRP-227-A") provides industry background listing of reactor internals components requiring inspection,

type of nondestructive examination (NDE) required for each component, timing for initial inspections, and criteria for evaluating inspection results. MRP-227-A provides a standardized approach to PWR internals aging management for each unique reactor design (Westinghouse, B&W, and CE).

- MRP-228, Revision 1 [15], "Inspection Standard for PWR Internals," provides guidance on the qualification/demonstration of the required NDE techniques and other criteria pertaining to the actual performance of the inspections.

The PWROG has developed WCAP-17096-NP, Revision 2, "Reactor Internals Acceptance Criteria Methodology and Data Requirements" for the MRP-227, Revision 0 components, where feasible [16]. This document has been submitted to the NRC for review and approval, and will be updated to incorporate changes from MRP-227-A [6]. Final reports are to be developed and available for industry use in support of planned license renewal inspection commitments. In some cases, individual plants will develop plant-specific acceptance criteria for some internals components where a generic approach is not practical.

The SGS reactor internals for SGS Unit 2 are integral with the reactor coolant system (RCS) of a Westinghouse four-loop nuclear steam supply system (NSSS). Illustrations of typical reactor internals are provided in Figures A-1 through A-14.

As described in the SGS license renewal application (LRA) [1], the reactor vessel internals consist of the upper core support structure, the lower core support structure, and the in-core instrumentation support structure, where each of these major components has a distinct purpose. Also included are the flux thimble tubes that extend from the penetrations on the reactor vessel lower head up to the seal table. In addition, the major structural welds that form or join the major structures, the minor structural welds joining parts such as lifting lugs, supports, and tubes to the major structures, and the fasteners and alignment pins that guide, align, and fasten the major structures are within the scope of the reactor vessel internals. The reactor vessel internals also include the fuel assemblies and the rod cluster control assemblies that are supported by all three structures; however, these are subject to replacement in accordance with the Reload Control Process and as such, they are short-lived components and are not subject to the aging management requirements of MRP-227-A.

The upper core support structure consists of the upper support assembly, the upper core plate, support columns, and the control rod guide assemblies. The support columns establish the spacing between the upper support assembly and the upper core plate. The upper core plate consists of openings for the control rod guide tubes, and for the distribution of reactor coolant flow via orifice plates, integral flow mixers, and open holes. The control rod guide tube assemblies shield and guide the control rod drive shafts and control rods. They are fastened to the upper core plate and are guided by pins into the upper core plate for proper orientation and support. A large circumferential hold down spring restrains axial movements. The entire upper core support structure is removed as a unit during refueling operations to permit access to the fuel assemblies.

The purpose of the upper core support structure is to contain the guide tube assemblies that shield and guide the control rod drive shafts and control rods. This structure engages the top of the fuel assemblies and provides structural support experienced by transverse loadings from coolant crossflow and other

design conditions. The upper core support structure also provides structural support for vertical loads from the fuel, hydraulic forces, control rod dynamics, and other design loadings.

The lower core support structure remains in place in the reactor vessel during most refueling operations and is only removed to perform scheduled inspections. The lower core support structure is supported at its upper flange from a ledge in the reactor vessel and its lower end is restrained from transverse motion by a radial support system attached to the reactor vessel wall. The lower core support structure consists of the core barrel, the core baffles, the flow distribution (diffuser) plate, the lower core plate and support columns, the thermal shield, and the core support forging, which is welded to the core barrel. The core barrel supports and contains the fuel assemblies. The core barrel directs coolant flow upwards through the reactor vessel by means of the bottom-mounted flow distribution plate and the core baffles. The lower core support plate provides support for the support columns, reactor coolant flow distribution, and support and orientation of the fuel assemblies. The one-piece thermal shield provides neutron shielding when fuel is present in the core, and is fixed to the core barrel at the top with rigid bolted connections. The bottom of the thermal shield is connected to the core barrel by means of axial flexures. This bottom support allows for differential axial growth of the shield/core barrel but restricts radial or horizontal movement of the bottom of the thermal shield. Specimen guides are welded to the thermal shield and allow for irradiation of test samples during operations. The core support is contoured to the bottom of the reactor vessel and receives the weight, hydraulic, and control rod dynamic loadings.

The purpose of the lower core support structure is to form a periphery enclosure of the core including core baffles and a bottom flow distribution plate for efficient flow distribution, provide neutron shielding by means of the thermal shield, and to provide structural support experienced by transverse loadings from coolant crossflow and other design conditions. The lower core support structure also provides structural support for vertical loads from the fuel, hydraulic forces, control rod dynamics, and other design loadings.

The purpose of the in-core instrumentation support structure is to provide structural support for the bottom-mounted in-core instrumentation (flux thimbles and thermocouples) and to maintain a pressure boundary between the reactor coolant and containment atmosphere.

The SGS LRA lists the following system intended functions for the reactor vessel internals [1]:

1. Maintain reactor core assembly geometry. The reactor vessel internals maintains core assembly geometry within the reactor to ensure core cooling, core reactivity control, and the integrity of the fuel cladding as a radioactive material barrier. 10 CFR 54.4(a)(1)

2. Achieve and maintain the reactor core subcritical for any mode of normal operation or event. The rod cluster control assemblies adjust the concentration of the neutron absorber in the core. 10 CFR 54.4(a)(1)

3. Introduce emergency negative reactivity to make the reactor subcritical. Following a reactor trip signal, all rod cluster control assemblies are released into the core to initiate a complete reactor trip. 10 CFR 54.4(a)(1)

4. Resist non-safety-related systems, structures, and components (SSCs) failure that could prevent satisfactory accomplishment of a safety-related function. The control rods are non-safety-related components that have the potential for spatial interactions with safety-related SSCs. 10 CFR 54.4(a)(2)

SGS Unit 2 was granted a license for extended operation by the NRC through the issuance of a SER in NUREG-2101 [3]. In the SER, the NRC concluded that the reactor vessel internal systems, structures, and components that are subject to an Aging Management Review (AMR) had been adequately identified, as required by 10 CFR 54.21(a)(3) and that the requirements of 10 CFR 54.29 [7] have been met. A listing of the SGS Unit 2 reactor vessel internals components and subcomponents that are subject to AMP requirements is included in Table B-1.

In accordance with 10 CFR Part 54 [7], frequently referred to as the License Renewal Rule, SGS has developed a PBD to manage the aging of reactor vessel internals components and structures in accordance with NUREG-1801, XI.M16 [8]. The U.S. nuclear industry, through the efforts of the MRP and PWROG, has further investigated the components and subcomponents that require aging management to support continued reliable function. As designated by the Nuclear Energy Institute (NEI) protocols in NEI 03-08 [17], "Guidelines for the Management of Materials Issues," each plant will be required to use MRP-227-A and MRP-228 to develop and implement an AMP for reactor internals no later than three years after the initial industry issuance of MRP-227, Revision 0. MRP-227, Revision 0 was issued in December 2008, and plant AMPs must therefore be completed by December 2011 or sooner, as required by plant-specific license renewal commitments. Revision 0 of this AMP was completed to satisfy this MRP-227 requirement. According to [4], SGS Unit 2 is a Category B plant that is expected to submit their RVI AMP/inspection plan based on the guidance of MRP-227-A, consistent with their commitments. Per the LRA [1], SGS Unit 2 has a commitment to submit an inspection plan for reactor internals for approval by the NRC no later than April 18, 2018.

The information contained in this AMP fully complies with the requirements and guidance of the referenced documents. The AMP will manage aging effects of the RVI so that the intended functions will be maintained consistent with the CLB for the period of extended operation.

3 PROGRAM OWNER

The PWR Vessel Internals Program [2] manages the effects of age-related degradation mechanisms of reactor vessel internals. The successful implementation and comprehensive long-term management of the SGS reactor vessel internals AMP will require the integration of PSEG Nuclear organizations, corporately and at SGS, and interaction with multiple industry organizations including, but not limited to, the ASME, MRP, NRC, and PWROG. The responsibilities of the individual PSEG Nuclear corporate and SGS groups are provided in the following paragraphs. PSEG Nuclear will maintain cognizance of industry activities related to PWR internals inspection and aging management, and will address/implement industry guidance stemming from those activities, as appropriate under NEI 03-08 practices.

The overall responsibility for administration of the reactor vessel internals AMP is SGS Unit 2 senior management.

Additional responsibilities and the appropriate responsible personnel, as described in Section 3 of [2], are discussed in the following subsections.

3.1 MANAGER OF ENGINEERING PROGRAMS

The manager of engineering programs is responsible for the overall implementation of the PWR Reactor Internals Inspection program.

3.2 SITE PROGRAM ENGINEER (SPE)

The SPE is responsible for executing the PWR Reactor Internals Inspection program per the requirements of MRP-227-A [6].

3.3 CORPORATE PROGRAM ENGINEER (CPE)

The CPE is responsible for governance and oversight of PWR Reactor Internals Inspection program.

3.4 NDE SERVICES INSPECTOR/EXAMINER

The NDE services inspector/examiner is responsible for inspection of PWR Reactor Internals per the requirements of MRP-227-A and MRP-228 [15].

4 DESCRIPTION OF SGS REACTOR INTERNALS AGING MANAGEMENT PROGRAMS AND INDUSTRY PROGRAMS

The U.S. nuclear industry, through the combined efforts of utilities, vendors, and independent consultants, has defined a generic guideline to assist utilities in developing reactor internals plant-specific AMPs based on inspection and evaluation. The intent of the SGS Unit 2 AMP is to ensure the long-term integrity and safe operation of the reactor internals components. SGS has developed this AMP in conformance with the 10 Generic Aging Lessons Learned (GALL) [18] attributes and MRP-227-A [6].

This reactor internals AMP utilizes a combination of prevention, mitigation, and condition monitoring. Where applicable, credit is taken for existing programs such as water chemistry [19, 20] and inspections prescribed by the ASME Section XI ISI Program [10], and past and future mitigation projects such as control rod guide tube support pin replacement [38] and flux thimble tube replacement and inspection [39]. These existing programs are augmented with the inspections and evaluations recommended by MRP-227-A.

Aging degradation mechanisms that impact internals have been identified and documented in the LRA submitted by PSEG Nuclear [1]. The overall outcome of the reviews and the additional work performed by the industry, as summarized in MRP-227-A, is to provide appropriate augmented inspections for reactor internals components to provide early detection of the degradation mechanisms of concern. Therefore, this AMP is consistent with the existing SGS Unit 2 reactor vessel internals AMR methodology [8] and the additional industry work summarized in MRP-227-A. All sources are consistent and address concerns about component degradation resulting from the following eight material aging degradation mechanisms identified as affecting reactor internals:

- <u>Stress Corrosion Cracking</u>

  Stress corrosion cracking (SCC) refers to local, non-ductile cracking of a material due to a combination of tensile stress, environment, and metallurgical properties. In primary water, this may be referred to as PWSCC. The actual mechanism that causes SCC or PWSCC involves a complex interaction of environmental and metallurgical factors. The aging effect is cracking.

- <u>Irradiation-Assisted Stress Corrosion Cracking</u>

  Irradiation-assisted stress corrosion cracking (IASCC) is a unique form of SCC that occurs only in highly-irradiated components. The aging effect is cracking.

- <u>Wear</u>

  Wear is caused by the relative motion between adjacent surfaces, with the extent determined by the relative properties of the adjacent materials and their surface condition. The aging effect is loss of material.

- Fatigue

Fatigue is defined as the structural deterioration that can occur as the result of repeated stress/strain cycles caused by fluctuating loads and temperatures. After repeated cyclic loading of sufficient magnitude, microstructural damage can accumulate, leading to macroscopic crack initiation at the most highly affected locations. Subsequent mechanical or thermal cyclic loading can lead to growth of the initiated crack. Corrosion fatigue is included in the degradation description.

Low-cycle fatigue is defined as cyclic loads that cause significant plastic strain in the highly stressed regions, where the number of applied cycles is increased to the point where the crack eventually initiates. When the cyclic loads are such that significant plastic deformation does not occur in the highly stressed regions, but the loads are of such increased frequency that a fatigue crack eventually initiates, the damage accumulated is said to have been caused by high-cycle fatigue. The aging effects of low-cycle fatigue and high-cycle fatigue are additive.

Fatigue crack initiation and growth resistance are governed by a number of material, structural, and environmental factors such as stress range, loading frequency, surface condition, and presence of deleterious chemical species. Cracks typically initiate at local geometric stress concentrations such as notches, surface defects, and structural discontinuities. The aging effect is cracking.

- Thermal Aging Embrittlement

Thermal aging embrittlement is the exposure of delta ferrite within cast austenitic stainless steel (CASS) and precipitation-hardening (PH) stainless steel to high inservice temperatures, which can result in an increase in tensile strength, a decrease in ductility, and a loss of fracture toughness. Some degree of thermal aging embrittlement can also occur at normal operating temperatures for CASS and PH stainless steel internals. CASS components have a duplex microstructure and are particularly susceptible to this mechanism. While the initial aging effect is loss of ductility and toughness, unstable crack extension is the eventual aging effect if a crack is present and the local applied stress intensity exceeds the reduced fracture toughness.

- Irradiation Embrittlement

Irradiation embrittlement is also referred to as neutron embrittlement. When exposed to high-energy neutrons, the mechanical properties of stainless steel and nickel-based alloys can be changed. Such changes in mechanical properties include increasing yield strength, increasing ultimate strength, decreasing ductility, and a loss of fracture toughness. The irradiation embrittlement aging mechanism is a function of both temperature and neutron fluence. While the initial aging effect is loss of ductility and toughness, unstable crack extension is the eventual aging effect if a crack is present and the local applied stress intensity exceeds the reduced fracture toughness.

- Void Swelling and Irradiation Growth

  Void swelling is defined as a gradual increase in the volume of a component caused by formation of microscopic cavities in the material. These cavities result from the nucleation and growth of clusters of irradiation-produced vacancies. Helium produced by nuclear transmutations can have a significant impact on the nucleation and growth of cavities in the material. Void swelling may produce dimensional changes that exceed the tolerances on a component. Strain gradients produced by differential swelling in the system may produce significant stresses. Severe swelling (>5 percent by volume) has been correlated with extremely low fracture toughness values. Also included in this mechanism is irradiation growth of anisotropic materials, which is known to cause significant dimensional changes within in-core instrumentation tubes that are fabricated from zirconium alloys. While the initial aging effect is dimensional change and distortion, severe void swelling may result in cracking under stress.

- Thermal and Irradiation-Enhanced Stress Relaxation or Irradiation-Enhanced Creep

  The loss of preload aging effect can be caused by the aging mechanisms of stress relaxation or creep. Thermal stress relaxation (or primary creep) is defined as the unloading of preloaded components due to long-term exposure to elevated temperatures, as seen in PWR internals. Stress relaxation occurs under conditions of constant strain where part of the elastic strain is replaced with plastic strain. Available data show that thermal stress relaxation appears to reach saturation in a short time (< 1000 hours) at PWR internals temperatures.

  Creep (or more precisely, secondary creep) is a slow, time- and temperature-dependent, plastic deformation of materials that can occur when subjected to stress levels below the yield strength (elastic limit). Creep occurs at elevated temperatures where continuous deformation takes place under constant strain. Secondary creep in austenitic stainless steels is associated with temperatures higher than those relevant to PWR internals even after taking into account gamma heating. However, irradiation-enhanced creep (or more simply, irradiation creep) or irradiation-enhanced stress relaxation (ISR) is an athermal process that depends on the neutron fluence and stress, and it can also be affected by void swelling should it occur. The aging effect is a loss of mechanical closure integrity (or, preload) that can lead to unanticipated loading that, in turn, may eventually cause subsequent degradation by fatigue or wear and result in cracking.

The SGS Unit 2 RVI AMP is focused on meeting the requirements of the 10 elements of an aging management program as described in NUREG-1801, GALL Report Section XI.M16A for PWR Vessel Internals. In the SGS Unit 2 RVI AMP, this is demonstrated through application of existing SGS AMR methodology that credits inspections prescribed by the ASME Code Section XI ISI Program, existing SGS programs, and additional augmented inspections based on MRP-227-A recommendations. A description of the applicable existing SGS programs and compliance with the elements of the GALL is contained in the following subsections.

4.1 SGS PROGRAMS

SGS's overall strategy for managing aging in reactor internals components is supported by the following existing programs [1]:

- Primary Water Chemistry Program [19, 20]
- ASME Section XI ISI Program [10]
- Metal Fatigue of Reactor Coolant Pressure Boundary [40]

These are established programs that support the aging management of RCS components in addition to the reactor vessel internals components. Although affiliated with and supporting the reactor vessel internals AMP, these programs will continue to be managed under the existing structure.

SGS's overall strategy for managing aging in reactor internals components will also be supported by the new reactor vessel internals aging management activities (see Section 4.2.2), flux thimble tube inspections, and the metal fatigue of reactor pressure boundary program.

Brief descriptions of the programs are included in the following subsections.

4.1.1 Primary Water Chemistry Program

The SGS Unit 2 Primary Strategic Water Chemistry Plan is an existing program [19, 20] that provides activities for monitoring and controlling the chemical environments of the SGS primary cycle systems such that aging effects of system components are minimized. This program manages the aging effects of cracking, loss of material, reduction of neutron-absorbing capacity, and reduction of heat transfer. The program mitigates damage caused by corrosion and SCC and other aging mechanisms. This program includes provisions specified by NUREG-1801 for the verification of proper chemistry control and aging management, such that the intended functions of plant components will be maintained during the period of extended operation for SGS Unit 2.

The SGS Unit 2 water chemistry aging management program includes periodic sampling of primary water for the known detrimental contaminants specified in the EPRI PWR water chemistry guidelines to maintain their concentrations below levels known to result in loss of material or cracking. Sampling frequencies and action limits for each control parameter are defined in SGS-specific procedures.

SGS follows the guidance set forth in the EPRI PWR Primary Water Chemistry Guidelines [13], which is referenced in NUREG-1801, XI.M2 (which refers to Revision 3 of the guidelines - EPRI TR-105714). Later revisions of the guidelines will be used when issued. The limits imposed by the SGS Program meet the intent of the industry standard for addressing primary water chemistry [13] and includes SGS Unit 2 Plant UFSAR [9] limits for specific chemical control parameters.

The evaluation of this program against the 10 attributes in the GALL for Program XI.M2 in support of the SGS LRA remains applicable.

4.1.2 ASME Section XI Inservice Inspection Program

The ASME Code Section XI Inservice Program is part of the SGS ISI Program [10]. This existing program includes inspections that are performed to manage aging effects such as cracking, loss of fracture toughness and loss of material in Class 1, 2, and 3, piping and components exposed to air, reactor coolant, steam, treated water and treated borated water environments within the scope of license renewal. The SGS ASME Section XI ISI Program is augmented, as identified in the SGS LRA [1], to also manage effects of aging by other programs. For SGS Unit 2, inspections conducted under the reactor vessel

internals AMP will be controlled as a combination of ASME Code Section XI ISI examinations on core support structures and augmented examinations performed under the ISI Program for the reactor vessel internals components addressed within MRP-227-A. ASME Code Section XI, 10-year ISI examinations supporting the period of extended operation are currently scheduled for the 2RF25 (Fall 2021).

The evaluation of this program against the 10 attributes in the GALL for Program XI.M1 in support of the SGS LRA remains applicable.

4.1.3 Metal Fatigue of Reactor Pressure Boundary Program

The SGS Metal Fatigue of Reactor Pressure Boundary Program [40] is an existing program that manages cumulative fatigue usage for the reactor vessel, the pressurizer, the steam generators, Class 1 and non-Class 1 piping, and Class 1 components subject to the reactor coolant, treated borated water, and treated water environments. The Metal Fatigue of Reactor Pressure Boundary Program is a preventive program that monitors and tracks the number of critical thermal and pressure transients to ensure that the cumulative usage factors for selected reactor coolant system (RCS) components remain less than 1.00 through the period of extended operation. The program determines the number of transients that occur and updates the 60-year projections as required on an annual basis.

The evaluation of this program against the 10 attributes in the GALL for Program X.M1 in support of the SGS LRA remains applicable.

4.2 SUPPORTING SALEM NUCLEAR GENERATING STATION UNIT 2 PROGRAMS AND AGING MANAGEMENT SUPPORTIVE PLANT ENHANCEMENTS

4.2.1 Reactor Internals Aging Management Review Process

A comprehensive review of aging management of SGS Unit 2 reactor internals was performed according to the requirements of the License Renewal Rule [7]. This review was conducted in support of the SGS LRA [1]. The SGS Unit 2 LRA was approved by the NRC in NUREG-2101 [3]. The SGS LRA, subsection 2.3.1.3, Table 3.1.1 and Table 3.1.2-3 identified the reactor vessel internals components that are subject to AMR. The LRA Table 3.1.2-3 Reactor Vessel Internals – Summary of Aging Management Evaluation provides the detailed results of the AMR conducted on the reactor vessel internals components. It also includes a comparison to the relevant NUREG-1801, Volume 2 item to note consistencies and there are footnotes to explain exceptions.

The AMR supported the SGS LRA as follows:

1. Identified applicable aging effects requiring management.

2. Evaluated existing aging management programs and commitments to ensure that they adequately manage those aging effects.

3. Identified actions for modifications to existing programs or actions to create new aging management programs, and any other actions required to support the conclusions reached in the review.

AMRs were performed for each SGS Unit 2 system that contained long-lived, passive components requiring AMR and the results are incorporated into the SGS LRA. This review is not repeated here, but the results are fully incorporated into the SGS Unit 2 RVI AMP. It should be noted that some RVI components listed in the SGS Unit 2 AMR were not plant specific. The non-applicable components have been noted within Table B-1.

4.2.2 Pressurized Water Reactor Internals Aging Management Program

The PWR Internals aging management program [2] is a new program that provides reasonable assurance that the changes in dimensions, cracking, loss of fracture toughness, and loss of preload aging effects is adequately managed so that the intended functions of components within the scope of license renewal is maintained consistent with the current licensing basis during the period of extended operation.

The procedure [2] establishes the PWR Vessel Internals aging management program for the inspection, repair, replacement, degradation evaluation, and mitigation of the PWR Internals, which ensures that MRP guidelines are met. The PWR Vessel Internals program establishes a framework and structure to existing PWR Vessel Internals efforts and implements the guidelines of MRP-227-A [6] and MRP-228 [15].

The reactor coolant system CASS components are maintained by inspecting and evaluating the extent of thermal aging embrittlement in accordance with the requirements of the ASME Boiler and Pressure Vessel Code, Section XI. The SGS Unit 2 PWR Vessel Internals Program will be used to manage the loss of fracture toughness due to thermal aging and neutron irradiation embrittlement of CASS reactor vessel internals components exposed to reactor coolant and neutron flux.

4.2.3 Flux Thimble Tube Inspection

Flux thimble tubes are long, slender, stainless steel tubes that are seal welded at one end with flux thimble tube plugs, which pass through the vessel penetration, through the lower internals assembly, and finally extend to the top of the fuel assembly. The bottom-mounted instrumentation (BMI) column assemblies provide a path for the flux thimbles into the core from the bottom of the vessel and protect the flux thimbles during operation of the reactor. The flux thimble provides a path for the neutron flux detector into the core and is subject to reactor coolant pressure on the outside and containment pressure on the inside.

The SGS Flux Thimble Tube Inspection Program [39] is a new program that manages the loss of material due to wear of the flux thimble tube materials. It implements the recommendations of NRC Bulletin 88-09 [21] that a tube wear inspection procedure be established and maintained for Westinghouse-supplied reactors that use bottom-mounted flux thimble tube instrumentation. The program utilizes an inspection methodology such as ECT to inspect the flux thimble tubes on a periodic frequency to monitor wall thinning and predict when tubes would require repair or replacement. The program implements a wall thickness trending report.

The Flux Thimble Tube Inspection program establishes appropriate acceptance criteria (percent through-wall wear), based on industry guidance and including margin to include allowances for factors such as instrument uncertainty, uncertainties in wear scar geometry, and other potential inaccuracies, as applicable, to the inspection methodology.

Previously, a Flux Thimble Tube Inspection was in effect from 1985 to 1993. During the 2RF2 (1984) at SGS Unit 2, ECT was used to inspect the flux thimble tubes. Possible external damage/wall losses were observed on 16 tubes where they pass through the lower core support. In the subsequent outage (1986), ECT was used and the results indicated wall losses of over 40% for three flux thimble tubes. These three tubes were isolated. During the 2RF4 (1990) outage, SGS Unit 2 replaced all of the flux thimble tubes with an improved design. Examination of replaced flux thimble tubes (SGS Unit 1 during the 1993 outage) indicated that there was no significant wear on any of the inspected flux thimble tubes. Also, the examinations indicated that there was no cladding bulging or ovality detected. As a result of the 1993 examinations, SGS notified the NRC that it would discontinue future periodic inspections of the flux thimble tubes.

The re-implementation of the Flux Thimble Tube Inspection Program is consistent with the 10 elements of the aging management program XI.M37, Flux Thimble Tube Inspection, specified in NUREG-1801. The examples from the past program demonstrate the effectiveness of the program, where improved design changes were made to replace the flux thimble tube materials with those of an improved design and were later inspected to prove that there was minimal wear.

The evaluation of this program against the 10 attributes in the GALL for Program XI.M37 in support of the SGS LRA remains applicable.

4.2.4 Control Rod Guide Tube Support Pin Replacement Project

The control rod guide tube support pins are used to align the bottom of the control rod guide tube assembly into the top of the upper core plate. In general, SCC prevention is aided by adherence to strict primary water chemistry limits that effectively mitigate SCC and greatly reduce the probability of IASCC. The limits imposed by the Primary Water Chemistry Program at SGS Unit 2 are consistent with the latest EPRI guidelines as described in Section 4.1.

The original SGS Unit 2 support pins were fabricated from *INCONEL*® Alloy X-750 that was hot rolled, solution treated, and age hardened at various temperatures and times depending on heat, manufacturer, and fabrication date. Support pins made of this material with the associated heat treatments were shown to be susceptible to PWSCC and likely to fail during the lifetime of a nuclear power plant. Westinghouse developed an improved support pin design and fabrication technique that significantly reduced the susceptibility to PWSCC while maintaining the fatigue and wear requirements necessary to support continued uninterrupted service [22].

In response to the industry concern, the support pins were replaced at SGS Unit 2 during the 2RF12 (2002) outage; the replacement support pins utilized improved materials (strain-hardened 316 stainless steel) that support the proactive management of aging in reactor internals components. Detailed descriptions of the replacement are retained in the plant records [38].

4.2.5 Reactor Vessel Internals Fatigue Analyses

The SGS Unit 2 reactor vessel internals were implicitly designed for low-cycle fatigue based upon the reactor coolant system design transient projection for 40 years; this is identified as a time-limiting aging analysis (TLAA). Post-design analyses consist of two Westinghouse analyses; (1) a lower core plate evaluation based on the 1.4% uprate and (2) qualification of the SGS Unit 2 domed lower core support plate, also part of the 1.4% uprate project. The effect of the 1.4% uprate was deemed negligible; therefore, cumulative fatigue usage attributable to the thermal design transients did not change [1]. SGS Unit 2 monitors and counts the plant operational cycles [40]. If the 60-year projected number of cycles is less than the number of cycles used in the design fatigue analysis, then the fatigue analyses based upon the design transients will remain valid for 60 years of operation if the design transient severity also bounds the actual transient severity.

The calculations and transient monitoring program demonstrate that the effects of aging on the reactor vessel internals will be adequately managed for the period of extended operation in accordance with 10 CFR 54.21(c)(1)(i).

4.3 INDUSTRY PROGRAMS

4.3.1 MRP-227-A, Reactor Internals Inspection and Evaluation Guidelines

MRP-227-A [6], as discussed in Section 2, was developed by a team of industry experts including utility representatives, NSSS vendors, independent consultants, and international committee representatives who reviewed available data and industry experience on materials aging. The objective of the group was to develop a consistent, systematic approach for identifying and prioritizing inspection and evaluation requirements for reactor internals. The following subsections briefly describe the industry process.

4.3.1.1 MRP-227-A Reactor Vessel Internals Component Categorizations

MRP-227-A used a screening and ranking process to aid in the identification of required inspections for specific RVI components. MRP-227-A credited existing component inspections, when they were deemed adequate, as a result of detailed expert panel assessments conducted in conjunction with the development of the industry document. Through the elements of the process, the reactor internals for all currently licensed and operating PWR designs in the U.S. were evaluated in the MRP Program and appropriate inspection, evaluation, and implementation requirements for reactor internals were defined.

Based on the completed evaluations, the RVI components are categorized within MRP-227-A as "Primary" components, "Expansion" components, "Existing Programs" components, or "No Additional Measures" components, described as follows:

- Primary

  Those PWR internals that are highly susceptible to the effects of at least one of the eight aging mechanisms were placed in the Primary group. The aging management requirements that are needed to ensure functionality of Primary components are described in the I&E guidelines. The Primary group also includes components that have shown a degree of tolerance to a specific aging

degradation effect, but for which no highly susceptible component exists or for which no highly susceptible component is accessible.

- Expansion

    Those PWR internals that are highly or moderately susceptible to the effects of at least one of the eight aging mechanisms, but for which functionality assessment has shown a degree of tolerance to those effects, were placed in the Expansion group. The schedule for implementation of aging management requirements for Expansion components depends on the findings from the examinations of the Primary components at individual plants.

- Existing Programs

    Those PWR internals that are susceptible to the effects of at least one of the eight aging mechanisms and for which generic and plant-specific existing AMP elements are capable of managing those effects, were placed in the Existing Programs group.

- No Additional Measures Programs

    Those PWR internals for which the effects of all eight aging mechanisms are below the screening criteria were placed in the No Additional Measures group. Additional components were placed in the No Additional Measures group as a result of a failure mode, effects, and criticality analysis (FMECA) and the functionality assessment. No further action is required by these guidelines for managing the aging of the No Additional Measures components.

The categorization and analysis used in the development of MRP-227-A are not intended to supersede any ASME B&PV Code Section XI [5] requirements. Any components that are classified as core support structures, as defined in ASME B&PV Code Section XI IWB 2500, Category B-N-3, have requirements that remain in effect and may only be altered as allowed by 10 CFR 50.55a.

4.3.1.2 NEI 03-08 Guidance within MRP-227-A

The industry program requirements of MRP-227-A are classified in accordance with the requirements of the NEI 03-08 protocols. The MRP-227-A guideline includes Mandatory and Needed elements as follows:

- **Mandatory**

There is one Mandatory element:

1. *Each commercial U.S. PWR unit shall develop and document a program for management of aging of reactor internals components within thirty-six months following issuance of MRP-227-Rev. 0 (that is, no later than December 31, 2011).*

    SGS Unit 2 Applicability: MRP-227, Revision 0 was officially issued by the industry in December 2008. An AMP must therefore be developed by December 2011. PSEG Nuclear

developed procedure ER-AP-333, "Pressurized Water Reactor Vessel Internals Program," [2] and Revision 0 of this AMP to meet the commitment that is contained in MRP-227, Revision 0.

According to the NRC RIS [4], SGS Unit 2 qualifies as a Category B plant because they have a renewed license with a commitment to submit an AMP/inspection plan based on MRP-227-A but that have not yet been required to do so by their commitment. This AMP fulfills the license renewal commitment to submit an implementation schedule for SGS Unit 2 in accordance with MRP-227-A [6] to the NRC no later than April 18, 2018.

- **Needed**

There are five Needed elements, with the fifth element being conditional based on examination results:

1. *Each commercial U.S. PWR unit shall implement MRP-227-A, Tables 4-1 through 4-9 and Tables 5-1 through 5-3 for the applicable design within twenty-four months following issuance of MRP-227-A.*

   SGS Unit 2 Applicability: MRP-227-A augmented inspections will be incorporated into the SGS ISI for the period of extended operation. The applicable Westinghouse tables contained in MRP-227-A include Table 4-3 (Primary), Table 4-6 (Expansion), Table 4-9 (Existing), and Table 5-3 (Examination Acceptance and Expansion Criteria) and are attached herein as Appendix C, Tables C-1, C-2, C-3, and C-4 respectively.

2. *Examinations specified in the MRP-227-A guidelines shall be conducted in accordance with the Inspection Standard MRP-228* [15].

   SGS Unit 2 Applicability: Inspection standards will be in accordance with the requirements of MRP-228 [15]. These inspection standards will be used for augmented inspection at SGS Unit 2 as applicable where required by MRP-227-A directives.

3. *Examination results that do not meet the examination acceptance criteria defined in Section 5 of the MRP-22-A 7 guidelines shall be recorded and entered in the plant corrective action program and dispositioned.*

   SGS Unit 2 Applicability: SGS Unit 2 will comply with this requirement.

4. *Each commercial U.S. PWR unit shall provide a summary report of all inspections and monitoring, items requiring evaluation, and new repairs to the MRP Program Manager within 120 days of the completion of an outage during which PWR internals within the scope of MRP-227-A are examined.*

   SGS Unit 2 Applicability: As discussed in Section 4.3.3, PSEG Nuclear will participate in future industry efforts and will adhere to industry directives for reporting, response, and follow-up.

5. *If an engineering evaluation is used to disposition an examination result that does not meet the examination acceptance criteria in Section 5, this engineering evaluation shall be conducted in accordance with a NRC-approved evaluation methodology.*

SGS Unit 2 Applicability: SGS Unit 2 will evaluate any examination results that do not meet the examination acceptance criteria in Section 5 of MRP-227-A in accordance with an NRC-approved methodology.

4.3.1.3 GALL AMP Development Guidance

It should be noted that Section XI.M16A of NUREG-1801, Revision 2 [18] includes a description of the attributes that make up an acceptable AMP. These attributes are consistent with the SGS Unit 2 AMR process. Evaluation of the SGS Unit 2 RVI AMP against GALL attribute elements is provided in Section 5 of this AMP.

As part of its license renewal, PSEG Nuclear is committed to participate in industry activities associated with the development of the standard Industry Guideline for Inspection and Evaluation of Reactor Internals. The industry efforts have defined the required inspections and examination techniques for those components critical to aging management of RVI. The results of the industry recommended inspections, as published in MRP-227-A, serve as the basis for identifying any augmented inspections that are required to complete the SGS Unit 2 RVI AMP.

4.3.1.4 MRP-227-A Applicability to SGS Unit 2

The applicability of MRP-227-A to SGS Unit 2 requires compliance with the following MRP-227-A assumptions:

- *30 years of operation with high-leakage core loading patterns (fresh fuel assemblies loaded in peripheral locations) followed by implementation of a low-leakage fuel management strategy for the remaining 30 years of operation.*

  SGS Unit 2 Applicability: SGS Unit 2 fuel management program changed from a high- to a low-leakage core-loading pattern prior to 30 years of operation [41].

- *Base load operation, i.e., typically operates at fixed power levels and does not usually vary power on a calendar or load demand schedule.*

  SGS Unit 2 Applicability: SGS Unit 2 operates as a base load unit [41].

- *No design changes beyond those identified in general industry guidance or recommended by the original vendors.*

  SGS Unit 2 Applicability: MRP-227-A states that the recommendations are applicable to all U.S. PWR operating plants as of May 2007 for the three designs identified. SGS Unit 2 has not made any modifications to reactor internals components beyond those identified in general industry guidance or recommended by the original vendor since May 2007 [41].

Based on the applicability, as stated above, the MRP-227-A work is representative for SGS Unit 2.

4.3.2 WCAP-17451-P, Reactor Internals Guide Tube Wear

The PWROG recently funded a program to develop a tool to facilitate prediction of continued operation of reactor upper internals guide tubes from a guide card and lower guide tube continuous guidance wear standpoint, as well as to establish an initial inspection schedule based on the various guide tube designs for the utilities participating in this program. A technical basis document was created for this program, WCAP-17451-P, Revision 1, "Reactor Internals Guide Tube Wear – Westinghouse Domestic Fleet Operational Projections" [44] which developed a guide plate (card) initial inspection schedule for Westinghouse NSSS designed plants. Per EPRI interim guidance letter, MRP 2014-006 [48], MRP group members have endorsed the guide plate (card) inspection requirements to be adopted within the next revision to MRP-227-A [6].

SGS Unit 2 is a four loop plant with a 17x17 standard guide tube design. According to Section 5.4 of the WCAP-17451-P [44], the generic initial guide card and continuous guidance inspection measurement effective full-power years (EFPY) range for this guide tube design is 24 to 28 EFPY. SGS Unit 2 was evaluated as a part of this technical basis and therefore, an alternative initial inspection measurement can be performed during an outage within a time range from 24 to 32 EFPY.

4.3.3 Ongoing Industry Programs

The U.S. industry, through both the EPRI/MRP and the PWROG, continues to sponsor activities related to RVI aging management, including planned development of a standard NRC submittal template, development of a plant-specific implementation program template for currently licensed U.S. PWR plants, and development of acceptance criteria and inspection disposition processes. PSEG Nuclear will maintain cognizance of industry activities related to PWR internals inspection and aging management; and will address/implement industry guidance stemming from those activities, as appropriate under NEI 03-08 practices.

4.4 SUMMARY

It should be noted that the PSEG Nuclear, the MRP, and the PWROG approaches to aging management are based on the GALL approach to aging management strategies. This approach includes a determination of which reactor internals passive components are most susceptible to the aging mechanisms of concern followed by determination of the proper inspection or mitigating program to provide reasonable assurance that the component will continue to perform its intended function through the period of extended operation. The GALL-based approach was used for the initial basis of the SGS LRA [1] that resulted in the NRC SER in NUREG-2101 [3].

The approach used to develop the SGS Unit 2 AMP is fully compliant with regulatory directives and approved documents. The additional evaluations and analysis completed by the MRP industry group have provided clarification on the level of inspection quality needed to determine the proper examination method and frequencies. The tables provided in MRP-227-A and included as Appendix C of this AMP provide the level of examination required for each of the components evaluated.

It is the PSEG Nuclear position that the SGS LRA, combined with any additional augmented inspections required by the MRP-227-A industry tables provided in Appendix C, provides reasonable assurance that

the reactor internals passive components will continue to perform their intended functions through the period of extended operation.

5 SALEM NUCLEAR GENERATING STATION REACTOR INTERNALS AGING MANAGEMENT PROGRAM ATTRIBUTES

The SGS Unit 2 RVI AMP is credited for aging management of RVI components for the following eight aging degradation mechanisms and their associated effects:

- Stress corrosion cracking
- Irradiation-assisted stress corrosion cracking
- Wear
- Fatigue
- Thermal aging embrittlement
- Irradiation embrittlement
- Void swelling and irradiation growth
- Thermal and irradiation-enhanced stress relaxation or irradiation-enhanced creep

The attributes of the SGS Unit 2 RVI AMP and compliance with NUREG-1801 (GALL Report), Section XI.M16A, "PWR Vessel Internals" [18] are described in this section. The GALL identifies 10 attributes for successful component aging management. The framework for assessing the effectiveness of the projected program is established by the use of the 10 elements of the GALL.

PSEG Nuclear fully utilized the GALL process contained in NUREG-1801, Revision 1 [42], in performing the AMR of the reactor internals in the license renewal process. However, PSEG Nuclear made a commitment [1, 3] to incorporate the following: (1) SGS Unit 2 will continue to participate in the industry programs for investigating and managing aging effects on reactor internals, (2) SGS Unit 2 will evaluate and implement the results of the industry programs as applicable to the reactor internals, and (3) upon completion of these programs, but not less than 24 months before entering the period of extended operation, SGS Unit 2 will submit an inspection plan for reactor internals to the NRC for review and approval. Augmented inspections, based on required program enhancements resulting from industry programs, will become part of the ASME B&PV Code, Section XI Program.

This AMP is consistent with that process, includes consideration of the augmented inspections identified in MRP-227-A, and fully meets the requirements of the commitment and GALL, Revision 2. Specific details of the SGS Unit 2 reactor internals AMP are summarized in the following subsections.

5.1 GALL REVISION 2 ELEMENT 1: SCOPE OF PROGRAM

GALL Report AMP Element Description

The scope of the program includes all RVI components at the SGS Unit 2 Nuclear Plant, which is built to a Westinghouse NSSS design. The scope of the program applies the methodology and guidance in the most recently NRC-endorsed version of MRP-227, which provides augmented inspection and flaw evaluation methodology for assuring the functional integrity of safety-related internals in commercial operating U.S. PWR nuclear power plants designed by B&W, CE, and Westinghouse. The scope of components considered for inspection under MRP-227 guidance includes core support structures (typically denoted as Examination Category B-N-3 by the ASME Code, Section XI), those RVI components that serve an intended license renewal safety function pursuant to criteria in 10 CFR 54.4(a)(1), and other RVI components whose failure could prevent

satisfactory accomplishment of any of the functions identified in 10 CFR 54.4(a)(1)(i), (ii), or (iii). The scope of the program does not include consumable items, such as fuel assemblies, reactivity control assemblies, and nuclear instrumentation, because these components are not typically within the scope of the components that are required to be subject to an aging management review (AMR), as defined by the criteria set in 10 CFR 54.21(a)(1). The scope of the program also does not include welded attachments to the internal surface of the reactor vessel because these components are considered to be ASME Code Class 1 appurtenances to the reactor vessel and are adequately managed in accordance with an applicant's AMP that corresponds to GALL AMP XI.M1, "ASME Code, Section XI Inservice Inspection, Subsections IWB, IWC, and IWD."

The scope of the program includes the response bases to applicable license renewal applicant action items (LRAAIs) on the MRP-227 methodology, and any additional programs, actions, or activities that are discussed in these LRAAI responses and credited for aging management of the applicant's RVI components. The LRAAIs are identified in the staff's safety evaluation on MRP-227 and include applicable action items on meeting those assumptions that formed the basis of the MRP's augmented inspection and flaw evaluation methodology (as discussed in Section 2.4 of MRP-227), and NSSS vendor-specific or plant-specific LRAAIs as well. The responses to the LRAAIs on MRP-227 are provided in Appendix C of the LRA.

The guidance of MRP-227 specifies applicability limitations to base-loaded plants and the fuel loading management assumptions upon which the functionality analyses were based. These limitations and assumptions require a determination of applicability by the applicant for each reactor and are covered in Section 2.4 of MRP-227 [18].

SGS Unit 2 Program Scope

The SGS Unit 2 reactor vessel internals consist of the upper core support structure, the lower core support structure, and the in-core instrumentation support structure, where each of these major components has a distinct purpose. Also included are the flux thimble tubes that extend from the penetrations on the reactor vessel lower head up to the seal table. The reactor vessel internals also include the fuel assemblies and the rod cluster control assemblies that are supported by all three structures. In addition, the major structural welds that form or join the major structures, the minor structural welds joining parts such as lifting lugs, supports, and tubes to the major structures, and the fasteners and alignment pins that guide, align, and fasten the major structures are within the scope of the reactor vessel internals. Additional reactor vessel internals details are provided in the SGS LRA [1].

The SGS Unit 2 reactor vessel internals subcomponents that require aging management review are indicated in Table 3.1.2-3 in the SGS LRA [1]. The table lists each subcomponent's intended function(s), material, and the aging effects that require management. A column in the tables lists the aging management program that is credited to address the component and aging effect during the period of extended operation. The NRC has reviewed and approved the aging management strategy presented in the Appendix B tables, as documented in the SER on license renewal [3].

The results of the industry research provided by MRP-227-A, summarized in the tables of Appendix C, provide the basis for the required augmented inspections, inspection techniques to permit detection and

characterizing of the aging effects (cracks, loss of material, loss of preload, etc.) of interest, prescribed frequency of inspection, and examination acceptance criteria. The information provided in MRP-227-A is rooted in the GALL methodology. The basic assumptions of MRP-227-A, Section 2.4 are met by SGS Unit 2 and are addressed in subsection 4.3.1.4 of this AMP. The Topical Report Conditions and Applicant/Licensee Action Items provided by the NRC in the Safety Evaluation (SE) on MRP-227, Revision 0 [6] are met by SGS and demonstration of compliance is addressed in Section 6.1 for the Topical Report Conditions and in Section 6.2 for the Applicant/Licensee Action Items. The SGS Unit 2 RVI AMP scope is additionally based on previously established and approved GALL Report approaches through application of the MRP-227-A [6] methodologies to determine those components that require aging management.

Conclusion

This element complies with the corresponding aging management attribute in NUREG-1801, Section XI.M16A [18] and Commitment 7 in the SER [3].

5.2 GALL REVISION 2 ELEMENT 2: PREVENTIVE ACTIONS

GALL Report AMP Element Description

> *The guidance in MRP-227 relies on PWR water chemistry control to prevent or mitigate aging effects that can be induced by corrosive aging mechanisms (e.g., loss of material induced by general, pitting corrosion, crevice corrosion, or stress corrosion cracking or any of its forms [SCC, PWSCC, or IASCC]). Reactor coolant water chemistry is monitored and maintained in accordance with the Water Chemistry Program. The program description, evaluation, and technical basis of water chemistry are presented in GALL AMP XI.M2, "Water Chemistry"* [18].

SGS Unit 2 Preventive Action

The SGS Unit 2 RVI AMP includes the Primary Water Chemistry Program [19, 20] as an existing program that complies with the requirements of this element. A description and applicability to the SGS Unit 2 RVI AMP is provided in the following subsection.

Primary Water Chemistry Program

To mitigate aging effects on component surfaces that are exposed to water as process fluid, chemistry programs are used to control water chemistry for impurities (e.g., dissolved oxygen, chloride, fluoride, and sulfate) that accelerate corrosion. This program relies on monitoring and control of water chemistry to keep peak levels of various contaminants below the system-specific limits. The SGS PWR Primary Water Chemistry Program [19, 20] is based on the current, approved revisions of EPRI PWR Primary Water Chemistry Guidelines [13].

This program is consistent with the corresponding program described in the GALL Report [18].

The limits of known detrimental contaminants imposed by the chemistry monitoring program are consistent with the EPRI PWR Primary Water Chemistry Guidelines [13].

Conclusion

This element complies with the corresponding aging management attribute in NUREG-1801, Section XI.M16A [18] and Commitment 7 in the SER [3].

5.3 GALL REVISION 2 ELEMENT 3: PARAMETERS MONITORED OR INSPECTED

GALL Report AMP Element Description

The program manages the following age-related degradation effects and mechanisms that are applicable in general to the RVI components at the facility: (a) cracking induced by SCC, PWSCC, IASCC, or fatigue/cyclical loading; (b) loss of material induced by wear; (c) loss of fracture toughness induced by either thermal aging or neutron irradiation embrittlement; (d) changes in dimension due to void swelling and irradiation growth, distortion, or deflection; and (e) loss of preload caused by thermal and irradiation-enhanced stress relaxation or creep. For the management of cracking, the program monitors for evidence of surface breaking linear discontinuities if a visual inspection technique is used as the non-destruction examination (NDE) method, or for relevant flaw presentation signals if a volumetric UT method is used as the NDE method. For the management of loss of material, the program monitors for gross or abnormal surface conditions that may be indicative of loss of material occurring in the components. For the management of loss of preload, the program monitors for gross surface conditions that may be indicative of loosening in applicable bolted, fastened, keyed, or pinned connections. The program does not directly monitor for loss of fracture toughness that is induced by thermal aging or neutron irradiation embrittlement, or by void swelling and irradiation growth; instead, the impact of loss of fracture toughness on component integrity is indirectly managed by using visual or volumetric examination techniques to monitor for cracking in the components and by applying applicable reduced fracture toughness properties in the flaw evaluations if cracking is detected in the components and is extensive enough to warrant a supplemental flaw growth or flaw tolerance evaluation under the MRP-227 guidance or ASME Code, Section XI requirements. The program uses physical measurements to monitor for any dimensional changes due to void swelling, irradiation growth, distortion, or deflection.

Specifically, the program implements the parameters monitored/inspected criteria for Westinghouse designed Primary Components in Table 4-3 of MRP-227. Additionally, the program implements the parameters monitored/inspected criteria for Westinghouse designed Expansion Components in Table 4-6 of MRP-227. The parameters monitored/inspected for Existing Program Components follow the bases for referenced Existing programs, such as the requirements for ASME Code Class RVI components in ASME Code, Section XI, Table IWB-2500-1, Examination Categories B-N-3, as implemented through the applicant's ASME Code, Section XI program, or the recommended program for inspecting Westinghouse-designed flux thimble tubes in GALL AMP XI.M37, "Flux Thimble Tube Inspection." No inspections, except for those specified in ASME Code, Section XI, are required for components that are identified as requiring "No Additional Measure," in accordance with the analyses reported in MRP-227 [18].

SGS Unit 2 Parameters Monitored or Inspected

The SGS Unit 2 AMP monitors, inspects, and/or tests for the effects of the eight aging degradation mechanisms on the intended function of the SGS Unit 2 PWR internals components through inspection and condition monitoring activities in accordance with the augmented requirements defined under industry directives as contained in MRP-227-A and ASME Section XI [5].

This AMP implements the requirements for the Primary Component inspections from Table 4-3 of MRP-227-A (included in Appendix C of this AMP as Table C-1), the Expansion Component inspections from Table 4-6 of MRP-227-A (included in Appendix C of this AMP as Table C-2), and the Existing Component inspections from Table 4-9 of MRP-227-A (included in Appendix C of this AMP as Table C-3). These tables contain requirements to monitor and inspect the RVI through the period of extended operation to address the effects of the eight aging degradation mechanisms. It is noted in Appendix C, Table C-1 that the PWROG has recently developed initial examination period requirements for guide plate (card) wear for Westinghouse NSSS designed plants [44] that will replace the current requirements in MRP-227-A [6].

For license renewal, the ASME Section XI Program [10] consists of periodic volumetric, surface, and/or visual examination of components for assessment, signs of degradation, and corrective actions. The requirements of MRP-227-A only augment and do not replace or modify the requirements of ASME Section XI. This program is consistent with the corresponding program described in the GALL Report [18].

Appendices B and C of this AMP provide a detailed listing of the components and subcomponents and the parameters monitored, inspected, and/or tested.

Conclusion

This element complies with or exceeds the corresponding aging management attribute in NUREG-1801, Section XI.M16A [18] and Commitment 7 in the SER [3].

5.4 GALL REVISION 2 ELEMENT 4: DETECTION OF AGING EFFECTS

GALL Report AMP Element Description

The detection of aging effects is covered in two places: (a) the guidance in Section 4 of MRP-227 provides an introductory discussion and justification of the examination methods selected for detecting the aging effects of interest; and (b) standards for examination methods, procedures, and personnel are provided in a companion document, MRP-228. In all cases, well-established methods were selected. These methods include volumetric UT examination methods for detecting flaws in bolting, physical measurements for detecting changes in dimension, and various visual (VT-3, VT-1, and EVT-1) examinations for detecting effects ranging from general conditions to detection and sizing of surface-breaking discontinuities. Surface examinations may also be used as an alternative to visual examinations for detection and sizing of surface-breaking discontinuities.

Cracking caused by SCC, IASCC, and fatigue is monitored/inspected by either VT-1 or EVT-1 examination (for internals other than bolting) or by volumetric UT examination (bolting). The VT-3 visual methods may be applied for the detection of cracking only when the flaw tolerance of the component or affected assembly, as evaluated for reduced fracture toughness properties, is known and has been shown to be tolerant of easily detected large flaws, even under reduced fracture toughness conditions. In addition, VT-3 examinations are used to monitor/inspect for loss of material induced by wear and for general aging conditions, such as gross distortion caused by void swelling and irradiation growth or by gross effects of loss of preload caused by thermal and irradiation-enhanced stress relaxation and creep.

In addition, the program adopts the recommended guidance in MRP-227 for defining the Expansion criteria that needed to be applied to inspections of Primary Components and Existing Requirement Components and for expanding the examinations to include additional Expansion Components. As a result, inspections performed on the RVI components are performed consistent with the inspection frequency and sampling bases for Primary Components, Existing Requirement Components, and Expansion Components in MRP-227, which have been demonstrated to be in conformance with the inspection criteria, sampling basis criteria, and sample Expansion criteria in Section A.1.2.3.4 of NRC Branch Position RLSB-1.

Specifically, the program implements the parameters monitored/inspected criteria and bases for inspecting the relevant parameter conditions for Westinghouse designed Primary Components in Table 4-3 of MRP-227 and for Westinghouse designed Expansion Components in Table 4-6 of MRP-227.

The program is supplemented by the following plant-specific Primary Component and Expansion Component inspections for the program (as applicable): for SGS Unit 2, no additional Primary or Expansion components are relevant to the scope of aging management for the RVI.

In addition, in some cases (as defined in MRP-227), physical measurements are used as supplemental techniques to manage for the gross effects of wear, loss of preload due to stress relaxation, or for changes in dimension due to void swelling, deflection or distortion. The physical measurements methods applied in accordance with this program include that for the hold down spring. The hold down spring at SGS Unit 2 is fabricated from Type 304 SS that requires inspection by physical measurement [18].

SGS Unit 2 Detection of Aging Effects

Detection of indications that are required by the ASME Code Section XI ISI Program [10] is well established and field-proven through the application of the Section XI ISI Program. Those augmented inspections that are taken from the MRP-227-A recommendations will be applied through use of the MRP-228 inspection standard. This AMP implements the augmented inspection requirements of Table 4-3, Table 4-6, and Table 4-9 from MRP-227-A for the Primary, Expansion, and Existing Components, respectively. These are included in Appendix C of this AMP for reference. These tables include the inspection frequency and sampling basis. For the Expansion Components of MRP-227-A, this AMP implements the expansion requirements of Table 5-3 of MRP-227-A (included in Appendix C of this AMP as Table C-4).

Inspection can be used to detect physical effects of degradation including cracking, fracture, wear, and distortion. The choice of an inspection technique depends on the nature and extent of the expected damage. The recommendations supporting aging management for the reactor internals, as contained in this report, are built around three basic inspection techniques: (1) visual, (2) ultrasonic, and (3) physical measurement. The three different visual testing (VT) and enhanced visual testing (EVT) techniques are VT-3, VT-1, and EVT-1. The assumptions and process used to select the appropriate inspection technique are described in the following subsections. Inspection standards developed by the industry for the application of these techniques for augmented reactor internals inspections are documented in MRP-228 [15].

VT-1 Visual Examinations

The acceptance criteria for visual examinations conducted under categories B-N-2 (welded core support structures and interior attachments to reactor vessels) and B-N-3 (removable core support structures) are defined in IWB-3520 [5]. VT-1 visual examination is intended to identify crack-like surface flaws. Unacceptable conditions for a VT-1 examination are:

- Crack-like surface flaws on the welds joining the attachment to the vessel wall that exceed the allowable linear flaw standards of IWB-3510 [5]
- Structural degradation of attachment welds such that the original cross-sectional area is reduced by more than 10 percent

These requirements are defined to ensure the integrity of attachment welds on the ferritic pressure vessel. Although the IWB-3520 criteria do not directly apply to austenitic stainless steel internals, the clear intent is to ensure that the structure will meet minimum flaw tolerance fracture requirements. In the MRP-227-A recommendations, VT-1 examinations have been identified for components requiring close visual examinations with some estimate of the scale of deformation or wear. Note that in MRP-227-A, VT-1 has only been selected to detect distortion as evidenced by small gaps between the upper-to-lower mating surfaces of CE-welded core shrouds assembled in two vertical sections. Therefore, no additional VT-1 inspections over and above those required by ASME Section XI ISI have been specified.

EVT-1 Enhanced Visual Examination for the Detection of Surface Breaking Flaws

In the augmented inspections detailed in MRP-227-A for reactor internals, the EVT-1 enhanced visual examination has been identified for inspection of components where surface-breaking flaws are a potential concern. Any visual inspection for cracking requires a reasonable expectation that the flaw length and crack mouth opening displacement meet the resolution requirements of the observation technique. The EVT-1 specification augments the VT-1 requirements to provide more rigorous inspection standards for stress corrosion cracking and has been demonstrated for similar inspections in boiling water reactor (BWR) internals. Enhanced visual examination (i.e., EVT-1) is also conducted in accordance with the requirements described for visual examination (i.e., VT-1) with additional requirements (such as camera scanning speed) currently being developed by the industry. Any recommendation for EVT-1 inspection will require additional analysis to establish flaw-tolerance criteria, which must take into account potential embrittlement due to thermal aging or neutron irradiation. The industry through the PWROG has developed an approach for acceptance criteria methodologies to

support plant-specific augmented examinations. This work is summarized in WCAP-17096-NP "Reactor Internals Acceptance Criteria Methodology and Data Requirements" [16]. The acceptance criteria developed using these methodologies may be created on either a generic or plant-specific basis because both loads and component dimensions may vary from plant to plant within a typical PWR design.

VT-3 Examination for General Condition Monitoring

In the augmented inspections detailed in the MRP-227-A for reactor internals, the VT-3 visual examination has been identified for inspection of components where general condition monitoring is required. The VT-3 examination is intended to identify individual components with significant levels of existing degradation. As the VT-3 examination is not intended to detect the early stages of component cracking or other incipient degradation effects, it should not be used when failure of an individual component could threaten either plant safety or operational stability. The VT-3 examination may be appropriate for inspecting highly redundant components (such as baffle-edge bolts), where a single failure does not compromise the function or integrity of the critical assembly.

The acceptance criteria for visual examinations conducted under categories B-N-2 (welded core support structures and interior attachments to reactor vessels) and B-N-3 (removable core support structures) are defined in IWB-3520. These criteria are designed to provide general guidelines. The unacceptable conditions for a VT-3 examination are:

- Structural distortion or displacement of parts to the extent that component function may be impaired;
- Loose, missing, cracked, or fractured parts, bolting, or fasteners;
- Foreign materials or accumulation of corrosion products that could interfere with control rod motion or could result in blockage of coolant flow through fuel;
- Corrosion or erosion that reduces the nominal section thickness by more than 5 percent;
- Wear of mating surfaces that may lead to loss of function;
- Structural degradation of interior attachments such that the original cross-sectional area is reduced more than 5 percent.

The VT-3 examination is intended for use in situations where the degradation is readily observable. It is meant to provide an indication of condition, and quantitative acceptance criteria are not generally required. In any particular recommendation for VT-3 visual examination, it should be possible to identify the specific conditions of concern. For instance, the unacceptable conditions for wear indicate wear that might lead to loss of function. Guidelines for wear in a critical-alignment component may be very different from the guidelines for wear in a large structural component.

Surface Examination

In order to further characterize discontinuities on the surface of components, surface examination can supplement either visual (VT-3) or (VT-1/EVT-1) examinations specified in the MRP-227-A guidelines. This supplemental examination may thus be used to reject or accept relevant indications. A surface examination is an examination that indicates the presence of surface discontinuities, and the ASME B&PV Code [5] lists magnetic particle, liquid penetrant, eddy current, and ultrasonic examination methods as surface examination alternatives. Here, only the electromagnetic testing (ET), also called eddy current surface examination method, is covered.

When selected for use as a supplemental examination to examinations performed in the MRP-227-A guidelines, an ET examination is conducted in accordance with the requirements of the inspection standard [15].

ET examination is widely used for heat exchanger tubing inspections. Eddy currents are induced in the inspected object by electromagnetic coils, with disruptions in the eddy current flow caused by surface or near-surface anomalies detected by suitable instrumentation. Industry experience with ET examination is relatively robust, especially in the aerospace and petroleum refinery industries. The experience base for PWR nuclear systems is moderately robust, in particular for examination of steam generator, flux thimble, and heat exchanger tubing.

Ultrasonic Testing

Volumetric examinations in the form of ultrasonic testing (UT) techniques can be used to identify and determine the length and depth of a crack in a component. Although access to the surface of the component is required to apply the ultrasonic signals, the flaw may exist in the bulk of the material. In this proposed strategy, UT inspections have been recommended exclusively for detection of flaws in bolts. For the bolt inspections, any bolt with a detected flaw should be assumed to have failed. The size of the flaw in the bolt is not critical because crack growth rates are generally high, and it is assumed that the observed flaw will result in failure prior to the next inspection opportunity. It has generally been observed through examination performance demonstrations that UT can reliably (90 percent or greater reliability) detect flaws that reduce the cross-sectional area of a bolt by 35 percent.

Failure of a single bolt does not compromise the function of the entire assembly. Bolting systems in the reactor internals are highly redundant. For any system of bolts, it is possible to demonstrate multiple acceptable bolting patterns. The evaluation program must demonstrate that the remaining bolts meet the requirements for an acceptable bolting pattern for continued operation. The evaluation procedures must also demonstrate that the pattern of remaining bolts contains sufficient margin such that continuation of the bolt failure rate will not result in failure of the system to meet the requirements for an acceptable bolting pattern before the next inspection.

Establishment of the acceptable bolting pattern for any system of bolts requires analysis to demonstrate that the system will maintain reliability and integrity in continuing to perform the intended function of the component. This analysis is highly plant-specific. Therefore, any recommendation for UT inspection of bolts assumes that the plant owner will work with the designer to establish acceptable bolting patterns prior to the inspection to support continued operation.

Physical Measurement Examination

Continued functionality can be confirmed by physical measurements to evaluate the impact caused by various degradation mechanisms such as wear or loss of functionality as a result of loss of preload or material deformation. For SGS Unit 2, direct physical measurements are required only for the hold down spring.

Conclusion

This element complies with or exceeds the corresponding aging management attribute in NUREG-1801, Section XI.M16A [18] and Commitment 7 in the SER [3].

5.5 GALL REVISION 2 ELEMENT 5: MONITORING AND TRENDING

GALL Report AMP Element Description

> *The methods for monitoring, recording, evaluating, and trending the data that result from the program's inspections are given in Section 6 of MRP-227 and its subsections. The evaluation methods include recommendations for flaw depth sizing and for crack growth determinations as well for performing applicable limit load, linear elastic and elastic-plastic fracture analyses of relevant flaw indications. The examinations and re-examinations required by the MRP-227 guidance, together with the requirements specified in MRP-228 for inspection methodologies, inspection procedures, and inspection personnel, provide timely detection, reporting, and corrective actions with respect to the effects of the age-related degradation mechanisms within the scope of the program. The extent of the examinations, beginning with the sample of susceptible PWR internals component locations identified as Primary Component locations, with the potential for inclusion of Expansion Component locations if the effects are greater than anticipated, plus the continuation of the Existing Programs activities, such as the ASME Code, Section XI, Examination Category B-N-3 examinations for core support structures, provides a high degree of confidence in the total program* [18].

SGS Unit 2 Monitoring and Trending

Operating experience with PWR reactor internals has been generally proactive. Flux thimble wear and control rod guide tube support pin cracking issues were identified by the industry and continue to be actively managed. The extremely low frequency of failure in reactor internals makes monitoring and trending based on OE somewhat impractical. The majority of the materials aging degradation models used to develop the MRP-227-A guidelines are based on test data from reactor internals components removed from service. The data is used to identify trends in materials degradation and forecast potential component degradation. The industry continues to share both material test data and OE through the auspices of the MRP and PWROG. PSEG Nuclear has in the past and will continue to maintain cognizance of industry activities and shared information related to PWR internals inspection and aging management as demonstrated in their quality programs [23, 24, 25, 26, 27, 28, 29, 30].

Nickel alloy reactor vessel internals components are included in the SGS PWR Vessel Internals program. The ASME Section XI ISI, Subsections IWB, IWC, and IWD, is used for monitoring and trending activities for loss of material and wear of nickel alloy reactor vessel internals.

Inspections credited in the SGS LRA are based on utilizing the 10-year ISI Program and the augmented inspections derived from MRP-227-A and repeated here in Appendix C. The MRP-227-A inspections only augment and do not replace the existing ASME Section XI ISI requirements. These inspections, where practical, are scheduled to be conducted in conjunction with typical 10-year ISI examinations.

Appendix C, Tables C-1, C-2, and C-3 identify the augmented Primary and Expansion inspection and monitoring recommendations, and the Existing programs credited for inspection and aging management. As discussed in MRP-227-A, inspection of the "Primary" components provides reasonable assurance for demonstrating component current capacity to perform the intended functions. It is noted in Appendix C, Table C-1 that the PWROG has recently developed initial examination period requirements for guide plate (card) wear for Westinghouse NSSS designed plants [44] that will replace the current requirements in MRP-227-A [6]. Table C-4 in Appendix C identifies the MRP-227-A expansion criteria from the Primary components. If these expansion criteria are met for a component, the associated Expansion component is to be inspected to manage the aging degradation.

Reporting requirements are included as part of the MRP-227-A guidelines. Consistent reporting of inspection results across all PWR designs will enable the industry to monitor reactor internals degradation on an ongoing industry basis as the period of extended operation moves forward. Reporting of examination results will allow the industry to monitor and trend results and take appropriate preemptive action through update of the MRP guidelines.

Conclusion

This element complies with or exceeds the corresponding aging management attribute in NUREG-1801, Section XI.M16A [18] and Commitment 7 in the SER [3].

5.6 GALL REVISION 2 ELEMENT 6: ACCEPTANCE CRITERIA

GALL Report AMP Element Description

Section 5 of MRP-227 provides specific examination acceptance criteria for the Primary and Expansion Component examinations. For components addressed by examinations referenced to ASME Code, Section XI, the IWB-3500 acceptance criteria apply. For other components covered by Existing Programs, the examination acceptance criteria are described within the Existing Program reference document.

The guidance in MRP-227 contains three types of examination acceptance criteria:

- *For visual examination (and surface examination as an alternative to visual examination), the examination acceptance criterion is the absence of any of the specific, descriptive relevant conditions; in addition, there are requirements to record and disposition surface breaking indications that are detected and sized for length by VT-1/EVT-1 examinations;*

- *For volumetric examination, the examination acceptance criterion is the capability for reliable detection of indications in bolting, as demonstrated in the examination Technical Justification; in addition, there are requirements for system-level assessment of bolted or pinned assemblies with unacceptable volumetric (UT) examination indications that exceed specified limits; and*

- *For physical measurements, the examination acceptance criterion for the acceptable tolerance in the measured differential height from the top of the plenum rib pads to the vessel seating surface in B&W plants are given in Table 5-1 of MRP-227. The acceptance criterion for physical measurements performed on the height limits of the Westinghouse-designed hold down springs are required for 304 SS hold down springs. SGS Unit 2 has a 304 SS hold down spring; therefore, SGS Unit 2 is required to produce acceptance criteria for the physical measurements on the hold down spring* [18]

SGS Unit 2 Acceptance Criteria

Those recordable indications that are the result of inspections required by the existing SGS ISI Program scope are evaluated in accordance with the applicable requirements of the ASME Code through the existing Corrective Action Program [23], inspection programs [10], and repair programs [31].

Inspection acceptance and expansion criteria are provided in Appendix C, Table C-4. These criteria will be reviewed periodically as the industry continues to develop and refine the information and will be included in updates to SGS procedures to enable the examiner to identify examination acceptance criteria considering state-of-the-art information and techniques.

Augmented inspections, as defined by the MRP-227-A requirements included in this AMP as Appendix C, Table C-1, Table C-2, and Table C-3, that result in recordable relevant conditions will be entered into the plant Corrective Action Program and addressed by appropriate actions that may include enhanced inspection, repair, replacement, mitigation actions, or analytical evaluations. An example of an analytical evaluation is using an acceptable bolting WCAP approach [32], such as those commonly used to support continued component or assembly functionality. Additional analysis to establish acceptable bolting pattern evaluation criteria for the baffle-former bolt assembly, as contained in various industry documents [32], is also considered in determining the acceptance of inspection results to support continued component or assembly functionality.

The industry, through various cooperative efforts, is working to construct a consensus set of tools in line with accepted and proven methodologies to support this element. One of these tools is the PWROG document WCAP-17096-NP, "Reactor Internals Acceptance Criteria Methodology and Data Requirements," [16], which details acceptance criteria methodology for the MRP-227 Primary and Expansion components. Status is monitored through direct PSEG Nuclear cognizance of industry (including PWROG) activities related to PWR internals inspection and aging management.

Conclusion

This element complies with or exceeds the corresponding aging management attribute in NUREG-1801, Section XI.M16A [18] and Commitment 7 in the SER [3].

5.7 GALL REVISION 2 ELEMENT 7: CORRECTIVE ACTIONS

GALL Report AMP Element Description

Corrective actions following the detection of unacceptable conditions are fundamentally provided for in each plant's corrective action program. Any detected conditions that do not satisfy the examination acceptance criteria are required to be dispositioned through the plant corrective action program, which may require repair, replacement, or analytical evaluation for continued service until the next inspection. The disposition will ensure that design basis functions of the reactor internals components will continue to be fulfilled for all licensing basis loads and events. Examples of methodologies that can be used to analytically disposition unacceptable conditions are found in the ASME Code, Section XI or in Section 6 of MRP-227. Section 6 of MRP-227 describes the options that are available for disposition of detected conditions that exceed the examination acceptance criteria of Section 5 of the report. These include engineering evaluation methods, as well as supplementary examinations to further characterize the detected condition, or the alternative of component repair and replacement procedures. The latter are subject to the requirements of the ASME Code, Section XI. The implementation of the guidance in MRP-227, plus the implementation of any ASME Code requirements, provides an acceptable level of aging management of safety-related components addressed in accordance with the corrective actions of 10 CFR Part 50, Appendix B or its equivalent, as applicable.

Other alternative corrective action bases may be used to disposition relevant conditions if they have been previously approved or endorsed by the NRC. Examples of previously NRC-endorsed alternative corrective actions bases include those corrective actions bases for Westinghouse-design RVI components that are defined in Tables 4-1, 4-2, 4-3, 4-4, 4-5, 4-6, 4-7 and 4-8 of Westinghouse Report No. WCAP-14577-Rev. 1-A, or for B&W-designed RVI components in B&W Report No. BAW-2248. Westinghouse Report No. WCAP-14577-Rev. 1-A was endorsed for use in an NRC SE to the Westinghouse Owners Group, dated February 10, 2001. B&W Report No. BAW-2248 was endorsed for use in an SE to Framatome Technologies on behalf of the B&W Owners Group, dated December 9, 1999. Alternative corrective action bases not approved or endorsed by the NRC will be submitted for NRC approval prior to their implementation [18].

SGS Unit 2 Corrective Action

The existing SGS Unit 2 procedure for Inservice Repair and Replacement [10, 31] and the established 10 CFR 50, Appendix B, Program [23, 33, 34] that addresses the elements of corrective actions, confirmation process, and administrative controls will be credited for this element. The Inservice Repair and Replacement procedure establishes the SGS Unit 2 repair and replacement requirements of ASME Code Section XI, "Rules for ISI of Nuclear Power Plant Components" [5]. These requirements include the identification of a repair cycle, repair plan, and verification of acceptability for replacements. SGS Unit 2 is committed to developing corrective actions for augmented inspections using repair and replacement procedures equivalent to those requirements in ASME B&PV Code, Section XI and MRP-227-A, Section 6 [6].

Conclusion

This element complies with the corresponding aging management attribute in NUREG-1801, Section XI.M16A [18] and Commitment 7 in the SER [3].

5.8 GALL REVISION 2 ELEMENT 8: CONFIRMATION PROCESS

GALL Report AMP Element Description

> *Site quality assurance procedures, review and approval processes, and administrative controls are implemented in accordance with the requirements of 10 CFR Part 50, Appendix B, or their equivalent, as applicable. It is expected that the implementation of the guidance in MRP-227 will provide an acceptable level of quality for inspection, flaw evaluation, and other elements of aging management of the PWR internals that are addressed in accordance with the 10 CFR Part 50, Appendix B, or their equivalent (as applicable), confirmation process, and administrative controls* [18].

SGS Unit 2 Confirmation Process

SGS has an established 10 CFR 50, Appendix B, Program [23, 33, 34] that addresses the elements of corrective actions, confirmation process, and administrative controls. The SGS Unit 2 Program includes non-safety-related structures, systems, and components. Quality assurance (QA) procedures, review and approval processes, and administrative controls are implemented in accordance with the requirements of 10 CFR 50, Appendix B.

Conclusion

This element complies with or exceeds the corresponding aging management attribute in NUREG-1801, Section XI.M16A [18] and Commitment 7 in the SER [3].

5.9 GALL REVISION 2 ELEMENT 9: ADMINISTRATIVE CONTROLS

GALL Report AMP Element Description

> *The administrative controls for such programs, including their implementing procedures and review and approval processes, are under existing site 10 CFR 50 Appendix B Quality Assurance Programs, or their equivalent, as applicable. Such a program is thus expected to be established with a sufficient level of documentation and administrative controls to ensure effective long-term implementation* [18].

SGS Unit 2 Administrative Controls

SGS Unit 2 has an established 10 CFR Part 50, Appendix B Program [23, 33, 34] that addresses the elements of corrective actions, confirmation process, and administrative controls. The SGS Unit 2 Program includes non-safety-related structures, systems, and components. QA procedures, review and

approval processes, and administrative controls are implemented in accordance with the requirements of 10 CFR 50, Appendix B.

Conclusion

This element complies with or exceeds the corresponding aging management attribute in NUREG-1801, Section XI.M16A [18] and Commitment 7 in the SER [3].

5.10 GALL REVISION 2 ELEMENT 10: OPERATING EXPERIENCE

GALL Report AMP Element Description

> *Relatively few incidents of PWR internals aging degradation have been reported in operating U.S. commercial PWR plants. A summary of observations to date is provided in Appendix A of MRP-227-A. The applicant is expected to review subsequent operating experience for impact on its program or to participate in industry initiatives that perform this function.*
>
> *The application of the MRP-227 guidance will establish a considerable amount of operating experience over the next few years. Section 7 of MRP-227 describes the reporting requirements for these applications, and the plan for evaluating the accumulated additional operating experience* [18].

SGS Unit 2 Operating Experience

Extensive industry and SGS Unit 2 OE has been reviewed during the development of the RVI AMP. The experience reviewed includes NRC Information Notices 84-18, "Stress Corrosion Cracking in PWR Systems" [35] and 98-11, "Cracking of Reactor Vessel Internal Baffle Former Bolts in Foreign Plants" [36]. Most of the industry operating experience reviewed has involved cracking of austenitic stainless steel baffle-former bolts, or SCC of high-strength internals bolting. SCC of control rod guide tube support pins has also been reported.

Early plant OE related to hot functional testing and reactor internals is documented in plant historical records. Inspections, performed as part of the 10-year ISI Program, have been conducted as designated by existing commitments and would be expected to discover overall general internals structure degradation. To date, very little degradation has been observed industry-wide.

Industry OE is routinely reviewed by PSEG Nuclear engineers using Institute of Nuclear Power Operations (INPO) OE, the Nuclear Network, and other information sources as directed under the SGS operating experience procedure [37], for the determination of additional actions and lessons learned. These insights, as applicable, can be incorporated into the plant systems quarterly health reports and further evaluated for incorporation into plant programs.

A review of industry and plant-specific experience with RVI reveals that the U.S. industry, including PSEG Nuclear and SGS Unit 2, has responded proactively to industry issues relative to reactor internals degradation. An example that demonstrates this proactive response is the replacement of control rod guide tube support pins at SGS Unit 2. Other relevant operating experience includes the SGS Unit 2 flux

thimble tube replacement and the experience gained through the conduct of ASME Section XI ISI. These are briefly described in the following paragraphs.

- SGS Unit 2 Control Rod Guide Tubes Support Pin Replacement

  The control rod guide tube support pins were replaced at SGS Unit 2 during the 2RF12 (2002) refueling outage (RO) [38]. The replacement pins included a material upgrade from X-750 to Type 316 stainless steel in support of managing aging in the component.

- SGS Unit 2 Flux Thimble Tubes Replacement

  As discussed in the SGS LRA [1], SGS Unit 2 replaced all of the flux thimble tubes during the 2RF4 (1990) outage with an improved design. The 1RF10 (1993) activities for SGS Unit 1 involved ECT of eleven of the improved design flux thimble tubes. The results indicated that there was no significant wear on any of the eleven inspected flux thimble tubes. Also, the examinations indicated that there was no cladding bulging or ovality detected.

- SGS Unit 2 ASME Section XI ISI, Subsection IWB, IWC, and IWD

  The operating experience of the ASME Section XI ISI, subsections IWB, IWC, and IWD program did not show any adverse trends in performance. Three specific examples of visual, ultrasonic, and dye penetrant examinations with indications [1] provide objective evidence that the effects of aging are effectively managed by the ASME Section XI ISI, subsections IWB, IWC, and IWD program.

A key element of the MRP-227-A guideline is the reporting of age-related degradation of RVI components. PSEG Nuclear, through its participation in PWROG and EPRI-MRP activities, will continue to benefit from the reporting of inspection information and will share its own OE with the industry through the reporting requirements of Section 7 of MRP-227-A. The collected information from MRP-227-A augmented inspections will benefit the industry in its continued response to RVI aging degradation.

Conclusion

This element complies with or exceeds the corresponding aging management attribute in NUREG-1801, Section XI.M16A [18] and Commitment 7 in the SER [3].

6 DEMONSTRATION

SGS has demonstrated a long-term commitment to aging management of reactor internals. This AMP is based on an established history of programs to identify and monitor potential aging degradation in the reactor internals. Programs and activities undertaken in the course of fulfilling that commitment include:

- The examinations required by ASME Section XI for the SGS Unit 2 reactor vessel internals have been performed during each 10-year interval since plant operations commenced.
- As documented in SGS Unit 2 operational procedures, Operating Experience Reports (OER) are continuously reviewed by SGS Unit 2 personnel for applicable issues that indicate a need for updated operating procedures or programs.
- Review of Nuclear Oversight Section (NOS) audit reports, NRC inspection reports, and INPO evaluations indicate no unacceptable issues related to RVI inspections [23, 33, 34].
- The Primary Water Chemistry Program at SGS Unit 2 has been effective in maintaining the levels of oxygen, halides, and sulfate sufficiently low to prevent SCC of the reactor vessel internals
- Replacement control rod guide tube support pins for SGS Unit 2 in 2002 were fabricated from Type 316 stainless steel (SS), which will provide additional resistance to PWSCC.
- SGS Unit 2 replaced all of the flux thimble tubes during the 2RF4 (1990) outage with an improved design.
- SGS has participated in the PWROG program to develop initial examination period requirements for guide plate (card) wear for Westinghouse NSSS designed plants [44].
- PSEG Nuclear has actively participated in past and ongoing EPRI and PWROG RVI activities. PSEG Nuclear will continue to maintain cognizance of industry activities related to PWR internals inspection and aging management and will address/implement the industry guidance, stemming from those activities as appropriate under NEI 03-08 practices.

This AMP fulfills the approved license renewal methodology requirement to identify the most susceptible components and to inspect those components with an indication detection level commensurate with the expected degradation mechanism indication. Augmented inspections derived from the information contained in MRP-227-A, the industry I&E Guidelines, have been utilized in this AMP to build on existing plant programs. This approach is expected to encourage detection of a degradation mechanism at its first appearance, which is consistent with the ASME Code approach to inspections. This approach provides reasonable assurance that the internals components will continue to perform their intended function through the period of extended operation.

Typical ASME Code Section XI examinations identified in the AMP for the period of extended operation are currently scheduled to be performed at SGS Unit 2 during the 2RF25 (2021). The augmented inspections discussed in compliance with MRP-227-A requirements have been integrated in the implementation schedule, which is shown in Section 7. Integration of the required inspections will be

tracked to completion. As discussed, the industry MRP-227-A guidelines also provide for updates as experience is gained through inspection results. This feedback loop will enable updates based on actual inspection experience.

The augmented inspections described in this document, as summarized in Appendix C, combined with the ASME Code Section XI ISI Program inspections, existing SGS Programs, and use of OERs, provide reasonable assurance that the reactor internals at SGS Unit 2 will continue to perform their intended functions through the period of extended operation.

Table 6-1 lists the seven topical report conditions and Section 6.2 lists the eight applicant action items that came out of the NRC review of MRP-227, as listed in [6], as well as their compliance within this AMP.

6.1 DEMONSTRATION OF TOPICAL REPORT CONDITIONS COMPLIANCE TO SE ON MRP-227, REVISION 0

Table 6-1 Topical Report Condition Compliance to SE on MRP-227

Topical Condition	Applicable/Not Applicable	Compliance in AMP
1. High consequence components in the "No Additional Measures" Inspection Category	Applicable	The upper core plate and the lower support forging or casting components are added to Table C-2 as "Expansion Components" linked to the "Primary Component," the control rod guide tube (CRGT) lower flange weld.
2. Inspection of components subject to irradiation-assisted stress corrosion cracking	Applicable	The upper and lower core barrel cylinder girth welds and the lower core barrel flange weld are moved from Table C-2 "Expansion Components" to Table C-1 "Primary Components."
3. Inspection of high consequence components subject to multiple degradation mechanisms	Not Applicable	Not applicable for SGS Unit 2 is a Westinghouse designed reactor.
4. Imposition of minimum examination coverage criteria for "Expansion" inspection category components	Applicable	Notes 2 through 4 were added to Table C-1, as well as Note 2 to Table C-2 to reflect this condition.
5. Examination frequencies for baffle-former bolts and core shroud bolts	Applicable	Not applicable for the core shroud bolts since SGS Unit 2 is a Westinghouse designed reactor. In Table C-1 for the baffle-former bolts, the inspection frequency was changed to be 10 years following the initial or baseline inspection, unless SGS Unit 2 provides an evaluation for NRC staff approval that justifies a longer interval between inspections. A note has been added in Table C-1 to address the inspection frequency for the baffle-former bolts.

Table 6-1 Topical Report Condition Compliance to SE on MRP-227

Topical Condition	Applicable/Not Applicable	Compliance in AMP
6. Periodicity of the re-examination of "Expansion" inspection category components	Applicable	The re-inspection frequency is 10 years following initial inspection, unless SGS Unit 2 provides an evaluation for NRC staff approval that justifies a longer interval between inspections. A note has been added in Table C-2 to address the inspection frequency for the expansion components.
7. Updating of MRP-227, Revision 0, Appendix A	Applicable	Section 5 is updated to reflect XI.M16A from GALL Revision 2 [18].

6.2 DEMONSTRATION OF APPLICANT/LICENSEE ACTION ITEM COMPLIANCE TO SE ON MRP-227, REVISION 0

6.2.1 SE Applicant/Licensee Action Item 1: Applicability of FMECA and Functionality Analysis Assumptions

As addressed in Section 3.2.5.1 of this SE, each applicant/licensee is responsible for assessing its plant's design and operating history and demonstrating that the approved version of MRP-227 is applicable to the facility. Each applicant/licensee shall refer, in particular, to the assumptions regarding plant design and operating history made in the FMECA and functionality analyses for reactors of their design (i.e., Westinghouse, CE, or B&W) which support MRP-227 and describe the process used for determining plant-specific differences in the design of their RVI components or plant operating conditions, which result in different component inspection categories. The applicant/licensee shall submit this evaluation for NRC review and approval as part of its application to implement the approved version of MRP-227. This is Applicant/Licensee Action Item 1 [6].

SGS Unit 2 Compliance

The process used to verify that SGS Unit 2 is reasonably represented by the generic industry program assumptions with regard to neutron fluence, temperature, materials, and stress values used in the development of MRP-227-A is as follows:

1. Identification of typical Westinghouse pressurized water reactor (PWR) internals components (MRP-191, Table 4-4 [14]).

2. Identification of SGS Unit 2 PWR internals components.

3. Comparison of the typical Westinghouse PWR internals components to the SGS Unit 2 PWR internals components.

    a. Confirmation that no additional items were identified by this comparison (primarily supports Applicant/Licensee Action Item (A/LAI) 2).

b. Confirmation that the materials identified for SGS Unit 2 are consistent with those materials identified in MRP-191, Table 4-4.

c. Confirmation that the SGS Unit 2 internals are the same as, or equivalent to, the typical Westinghouse PWR internals regarding design and fabrication.

4. Confirmation that the SGS Unit 2 operating history is consistent with the assumptions in MRP-227-A regarding core loading patterns.

5. Confirmation that the SGS Unit 2 RVI materials operated at temperatures within the original design basis parameters.

6. Determination of stress values based on design basis documents.

7. Confirmation that any changes to the SGS Unit 2 RVI components do not impact the application of the MRP-227-A generic aging management strategy.

SGS Unit 2 reactor internals components are reasonably represented by the design and operating history assumptions regarding neutron fluence, temperature, materials, and stress values in the MRP-191 generic FMECA and the MRP-232 functionality analyses based on the following.

1. SGS Unit 2 operating history is consistent with the assumptions in MRP-227-A with regard to neutron fluence, as stated in Section 4.3.1.4.

   a. The FMECA and functionality analyses for MRP-227-A were based on the assumption of 30 years of operation with high-leakage core loading patterns followed by 30 years of low-leakage core fuel management strategy. In Fuel Cycle C10 (1997) [41] at approximately 16 years of operation, SGS Unit 2 switched to use of a low-leakage core design. Therefore, SGS Unit 2 meets the fluence and fuel management assumptions in MRP-191 and requirements for MRP-227-A application.

   b. SGS Unit 2 has operated under base load conditions over the life of the plant [41]. Therefore, SGS Unit 2 satisfies the assumptions in MRP documents regarding operational parameters affecting fluence.

2. The SGS Unit 2 RVI operate between Thot and Tcold [41], which are not less than approximately 530.3°F for Tcold and not higher than 613.1°F for Thot. The design temperature for the reactor vessel is 650°F. SGS Unit 2 operating history is within original design basis parameters and therefore consistent with the assumptions used to develop the MRP-227-A aging management strategy with regard to temperature operational parameters.

3. SGS Unit 2 internals components and materials are comparable to the typical Westinghouse PWR internals components (MRP-191, Table 4-4).

   a. Some of the components listed in the AMR are not contained in SGS Unit 2. PSEG was being all-encompassing and conservative in creating the AMR by including

components that were clearly a part of the SGS Unit 2 reactor internals and components which were not confirmed as part of the internals [41]. The work done for responding to A/LAIs 1 and 2 confirmed the actual list of components and identified those that were not a part of the SGS Unit 2 reactor internals. This review identified one component in the AMR [1], listed as the Upper Internals Assembly: Capped Top Thermocouple Columns, which is not included in MRP-191. However, the review also determined that this component is not applicable to SGS Unit 2. Therefore, SGS meets the requirements for aging as outlined in MRP-227-A.

b. Materials identified for SGS Unit 2 are consistent or nearly equivalent with those materials identified in MRP-191, Table 4-4 for Westinghouse-designed plants [14], except for the guide plates/cards, and brackets, clamps, terminal blocks, and conduit straps (conduit support, conduit support gusset, gusset clamp, and thermocouple stop) which are identified as being or having the potential to be cast CF8. However, based on a review by an expert panel [45], it was determined that this has no effect on the recommended MRP aging management inspection sampling strategy. The guide cards/plates will remain classified as a "Primary" inspection item, and the brackets, clamps, terminal blocks and conduit straps will remain classified as "No Additional Measures" items based on a consideration of the likelihood of failure and likelihood of damage. There is no change in MRP-227-A inspection requirements as a result of the inclusion of CF8 for these components. In addition, several other components have different materials than specified in MRP-191, but these material differences have been determined to have no effect of the recommended MRP aging management inspection sampling strategy [45].

c. SGS Unit 2 internals are the same as, or equivalent to, the typical Westinghouse PWR internals regarding design and fabrication.

4. Modifications to the SGS Unit 2 reactor internals made over the lifetime of the plant are those specifically directed by Westinghouse, the Original Equipment Manufacturer (OEM) [41]. The design has been maintained over the lifetime of the plant as specified by the OEM, operational parameters are compliant with MRP-227-A requirements with regard to fluence and temperature, and the components and materials are the same as, or nearly equivalent to, those considered in MRP-191. Therefore, the SGS Unit 2 stress values are represented by the assumptions in MRP-191, MRP-232, and MRP-227-A, confirming the applicability of the generic FMECA.

Conclusion

The assumptions regarding operating history made in the FMECA and functionality analyses for the Westinghouse design apply to SGS Unit 2. There are no components at SGS Unit 2 not contained in the FMECA and functionality analysis. There are components with materials different than those assumed in the FMECA; however, evaluations have been completed to verify that these differences do not affect the current aging management strategy. SGS Unit 2 meets the requirements for application of MRP-227-A as a strategy for managing age-related material degradation in reactor internals components. SGS Unit 2

will implement and apply the approved version of MRP-227 (MRP-227-A) as a strategy for managing age-related material degradation in reactor internals components [6].

6.2.2 SE Applicant/Licensee Action Item 2: PWR Vessel Internal Components within the Scope of License Renewal

As discussed in Section 3.2.5.2 of this SE, consistent with the requirements addressed in 10 CFR 54.4, each applicant/licensee is responsible for identifying which RVI components are within the scope of LR for its facility. Applicants/licensees shall review the information in Tables 4-1 and 4-2 in MRP-189, Revision 1, and Tables 4-4 and 4-5 in MRP-191 and identify whether these tables contain all of the RVI components that are within the scope of LR for their facilities in accordance with 10 CFR 54.4. If the tables do not identify all the RVI components that are within the scope of LR for its facility, the applicant or licensee shall identify the missing component(s) and propose any necessary modifications to the program defined in MRP-227, as modified by this SE, when submitting its plant-specific AMP. The AMP shall provide assurance that the effects of aging on the missing component(s) will be managed for the period of extended operation. This issue is Applicant/Licensee Action Item 2 [6].

SGS Unit 2 Compliance

This A/LAI requires comparison of the RVI components that are within the scope of license renewal for SGS Unit 2 to those components contained in MRP-191, Table 4-4. A detailed tabulation of the SGS Unit 2 RVI components was completed and compared favorably to the typical Westinghouse PWR internals components in MRP-191. Review of the information in Tables 4-1 and 4-2 in MRP-189, Revision 1 was not necessary as SGS Unit 2 is a Westinghouse design reactor.

Several components have different materials than those specified in the MRP-191 assessment, but these have no effect on the recommended MRP aging; therefore, no modifications to the program details in MRP-227-A need to be proposed.

The guide plates/cards are considered in the MRP-191 assessment and identified as being 304 SS; however, for SGS Unit 2, the guide plates/cards have the potential to be cast CF8 material. An expert panel review was conducted, aligned with the process specified in MRP-191, Section 6 [14], and it was concluded with 100 percent consensus that there is no change to the current MRP conclusion that the guide plates/cards are a Primary inspection item [45]. Therefore, no modifications to the program detailed in MRP-227-A need to be proposed due to the assessment and screening of the RVI as developed in MRP-191 and MRP-232.

The brackets, clamps, terminal blocks and conduit straps (conduit support, conduit support gusset, gusset clamp, and thermocouple stop) are considered in the MRP-191 assessment and identified as being 304 SS; however, for SGS Unit 2, conduit support, conduit support gusset, gusset clamp, and thermocouple stop have the potential to be cast CF8 material. An expert panel review was conducted, aligned with the process specified in MRP-191, Section 6 [14], and it was concluded with 100 percent consensus that there is no change to the current MRP conclusion that the brackets, clamps, terminal blocks and conduit straps are not inspection components [45]. Therefore, no modifications to the program detailed in MRP-227-A

need to be proposed due to the assessment and screening of the RVI as developed in MRP-191 and MRP-232.

This supports the requirement that the AMP shall provide assurance that the effects of aging on the SGS Unit 2 RVI components within the scope of license renewal but not included in the generic Westinghouse-designed RVI components from Table 4-4 of MRP-191 will be managed for the period of extended operation.

The generic scoping and screening of the RVI as summarized in MRP-191 and MRP-232 to support the inspection sampling approach for aging management of reactor internals specified in MRP-227-A is applicable to SGS Unit 2 with no modifications.

Conclusion

All components required to be included in the SGS Unit 2 program are consistent with those contained in MRP-191. Several components have materials different than those specified in MRP-191; however, evaluations have been completed to show that these differences have no effect on the MRP aging management strategy. SGS Unit 2 meets the requirement for application of MRP-227-A as a strategy for managing age-related degradation in reactor internals components.

6.2.3 SE Applicant/Licensee Action Item 3: Evaluation of the Adequacy of Plant-Specific Existing Programs

> *As addressed in Section 3.2.5.3 in this SE, applicants/licensees of CE and Westinghouse are required to perform plant-specific analysis either to justify the acceptability of an applicant's/licensee's existing programs, or to identify changes to the programs that should be implemented to manage the aging of these components for the period of extended operation. The results of this plant-specific analyses and a description of the plant-specific programs being relied on to manage aging of these components shall be submitted as part of the applicant's/licensee's AMP application. The CE and Westinghouse components identified for this type of plant-specific evaluation include: CE thermal shield positioning pins and CE in-core instrumentation thimble tubes (Section 4.3.2 in MRP-227), and Westinghouse guide tube support pins (split pins) (Section 4.3.3 in MRP-227). This is Applicant/Licensee Action Item 3* [6].

SGS Unit 2 Compliance

SGS Unit 2 is compliant with the requirements in Table 4-9 of MRP-227-A, as shown in Appendix C, Table C-3. This is detailed in the plant-specific SGS program documents for ASME Section XI [10] and the plant-specific flux thimble program [39].

In response to the industry concern, the control rod guide tube support pins fabricated from *INCONEL*® Alloy X-750 were replaced at SGS Unit 2 during the 2RF12 (2002) outage; the replacement support pins utilized improved materials (strain-hardened 316 stainless steel) that support the proactive management of aging in reactor internals components. Detailed descriptions of the replacement are retained in the plant records [38].

Conclusion

SGS Unit 2 complies with Applicant/Licensee Action Item 3 of the NRC SE on MRP-227, Revision 0, and therefore meets the requirement for application of MRP-227-A as a strategy for managing age-related material degradation in reactor internals components.

6.2.4 SE Applicant/Licensee Action Item 4: B&W Core Support Structure Upper Flange Stress Relief

As discussed in Section 3.2.5.4 of this SE, the B&W applicants/licensees shall confirm that the core support structure upper flange weld was stress relieved during the original fabrication of the Reactor Pressure Vessel in order to confirm the applicability of MRP-227, as approved by the NRC, to their facility. If the upper flange weld has not been stress relieved, then this component shall be inspected as a "Primary" inspection category component. If necessary, the examination methods and frequency for non-stress relieved B&W core support structure upper flange welds shall be consistent with the recommendations in MRP-227, as approved by the NRC, for the Westinghouse and CE upper core support barrel welds. The examination coverage for this B&W flange weld shall conform to the staff's imposed criteria as described in Sections 3.3.1 and 4.3.1 of this SE. The applicant's/licensee's resolution of this plant-specific action item shall be submitted to the NRC for review and approval. This is Applicant/Licensee Action Item 4 [6].

SGS Unit 2 Compliance

This Applicant/Licensee Action Item is not applicable to SGS Unit 2 since it is a Westinghouse design reactor, and this item only applies to B&W plants.

Conclusion

Applicant/Licensee Action Item 4 of the NRC SE on MRP-227, Revision 0 is not applicable to SGS Unit 2.

6.2.5 SE Applicant/Licensee Action Item 5: Application of Physical Measurements as part of I&E Guidelines for B&W, CE, and Westinghouse RVI Components

As addressed in Section 3.3.5 in this SE, applicants/licensees shall identify plant-specific acceptance criteria to be applied when performing the physical measurements required by the NRC-approved version of MRP-227 for loss of compressibility for Westinghouse hold down springs, and for distortion in the gap between the top and bottom core shroud segments in CE units with core barrel shrouds assembled in two vertical sections. The applicant/licensee shall include its proposed acceptance criteria and an explanation of how the proposed acceptance criteria are consistent with the plants' licensing basis and the need to maintain the functionality of the component being inspected under all licensing basis conditions of operation during the period of extended operation as part of their submittal to apply the approved version of MRP-227. This is Applicant/Licensee Action Item 5 [6].

SGS Unit 2 Compliance

SGS Unit 2 utilizes a Type 304 SS hold down spring; therefore, PSEG Nuclear is planning to perform inspections/physical measurements on the SGS Unit 2 hold down spring according to MRP-227-A. The proposed acceptance criteria for the physical measurements have been completed and are consistent with the licensing basis for SGS Unit 2 [46].

Acceptance criteria for distortion in the gap between the top and bottom core shroud segments are not applicable to SGS Unit 2 since it is a Westinghouse design reactor, and this element only applies to CE plants with core barrel shrouds assembled in two vertical sections.

Conclusion

SGS Unit 2 complies with Applicant/Licensee Action Item 5 of the NRC SE on MRP-227, Revision 0, and therefore meets the requirement for application of MRP-227-A as a strategy for managing age-related material degradation in reactor internals components.

6.2.6 SE Applicant/Licensee Action Item 6: Evaluation of Inaccessible B&W Components

> *As addressed in Section 3.3.6 in this SE, MRP-227 does not propose to inspect the following inaccessible components: the B&W core barrel cylinders (including vertical and circumferential seam welds), B&W former plates, B&W external baffle-to-baffle bolts and their locking devices, B&W core barrel-to-former bolts and their locking devices, and B&W core barrel assembly internal baffle-to-baffle bolts. The MRP also identified that although the B&W core barrel assembly internal baffle-to-baffle bolts are accessible, the bolts are non-inspectable using currently available examination techniques.*
>
> *Applicants/licensees shall justify the acceptability of these components for continued operation through the period of extended operation by performing an evaluation, or by proposing a scheduled replacement of the components. As part of their application to implement the approved version of MRP-227, applicants/licensees shall provide their justification for the continued operability of each of the inaccessible components and, if necessary, provide their plan for the replacement of the components for NRC review and approval. This is Applicant/Licensee Action Item 6* [6].

SGS Unit 2 Compliance

This Applicant/Licensee Action Item is not applicable to SGS Unit 2 since it is a Westinghouse design reactor, and this item only applies to B&W plants.

Conclusion

Applicant/Licensee Action Item 6 of the NRC SE on MRP-227, Revision 0 is not applicable to SGS Unit 2.

6.2.7 SE Applicant/Licensee Action Item 7: Plant-Specific Evaluation of CASS Materials

As discussed in Section 3.3.7 of this SE, the applicants/licensees of B&W, CE, and Westinghouse reactors are required to develop plant-specific analyses to be applied for their facilities to demonstrate that B&W IMI guide tube assembly spiders and CRGT spacer castings, CE lower support columns, and Westinghouse lower support column bodies will maintain their functionality during the period of extended operation or for additional RVI components that may be fabricated from CASS, martensitic stainless steel or precipitation hardened stainless steel materials. These analyses shall also consider the possible loss of fracture toughness in these components due to thermal and irradiation embrittlement, and may also need to consider limitations on accessibility for inspection and the resolution/sensitivity of the inspection techniques. The requirement may not apply to components that were previously evaluated as not requiring aging management during development of MRP-227. That is, the requirement would apply to components fabricated from susceptible materials for which an individual licensee has determined aging management is required, for example during their review performed in accordance with Applicant/Licensee Action Item 2. The plant-specific analysis shall be consistent with the plant's licensing basis and the need to maintain the functionality of the components being evaluated under all licensing basis conditions of operation. The applicant/licensee shall include the plant-specific analysis as part of their submittal to apply the approved version of MRP-227. This is Applicant/Licensee Action Item 7 [6].

SGS Unit 2 Compliance

A/LAI 7, from the NRC's final SE on MRP-227, revision 0, states that, for assessment of CASS materials, the applicant/licensee for license renewal may apply the criteria detailed in [43] as the basis for determining whether the CASS materials are susceptible to the thermal aging mechanism. If the application of the applicable screening criteria for the component's material demonstrates that the components are not susceptible to either thermal embrittlement (TE) or irradiation embrittlement (IE), or to the synergistic effects of TE and IE combined, then no other evaluation would be necessary. The SGS Unit 2 CASS RVI components and the assessment of their susceptibility to TE are summarized in Table 6-2. The B&W and CE components fabricated of CASS material, as discussed in A/LAI 7, are not applicable to SGS Unit 2 since it is a Westinghouse design reactor.

Using the chemistry data from retrieved certified material test reports (CMTRs) as input into Hull's formula per the guidance of [47], the ferrite content was determined. Based on the criteria of [43], the SGS Unit 2 CASS (Grade CF8) upper support column - orifice bases, upper support column mixing bases, core support casting and (Grade CF8) lower support column caps are not susceptible to TE.

The lower internals assembly column cap is a CASS piece welded onto the top of the core support column shaft. These two pieces together constitute the lower internals assembly – column body. The development of the MRP-227-A aging management strategy considered the lower support column as one complete unit denoted as the "lower support column assemblies – lower support column bodies." Under the lower support column bodies in MRP-191, both Type 304 SS and CF8 CASS material were considered. Since the lower internals assembly column cap is part of the lower support column body, it was addressed by the generic industry FMECA and functionality analysis; therefore, it is subject to the same inspection

requirements as the lower support column bodies (cast). The inspection of the lower support column body includes the column cap.

As shown in Table 6-2, the remaining SGS Unit 2 components have Grade CF8 specified as either a primary material or an alternate material on their engineering drawings. Where alternate A351 Grade CF8 is permitted, in the evaluation supporting A/LAI 7, it is conservative to assume that the material used for these components is Grade CF8. Confirmation of material composition under TE susceptibility thresholds was not demonstrated for these remaining components; therefore, it is conservatively assumed that they are potentially susceptible to TE.

In the development of MRP-191, guide plates and upper instrumentation column supports were screened as wrought material (304 SS). The material difference for the SGS Unit 2 guide plates/cards and brackets, clamps, terminal blocks, and conduit straps (conduit support, conduit support gusset, gusset clamp, and thermocouple stop) is addressed in Sections 6.2.1 and 6.2.2.

Irradiation may also cause a material to undergo embrittlement. In MRP-191, the guide tube lower flange, mixing device, upper support column base, and lower support column each screened-in as susceptible to IE. The guide tube intermediate flange, guide plate, upper instrumentation supports, and core support casting screened below the MRP-191 fluence screening level; thus, they are not susceptible to IE [45].

No martensitic SS or martensitic PH-SS components were identified for the SGS Unit 2 RVI.

Conclusion

It is concluded that continued application of the MRP-227-A strategy will meet the requirement for managing age-related degradation of the SGS Unit 2 CASS RVI components.

Table 6-2 Summary of SGS Unit 2 CASS Components and their Susceptibility to TE

CASS Component	SGS RVI AMR Component Name [1]	Molybdenum Content	Casting	Ferrite Content	Susceptibility to TE (Based on the NRC Criteria [43])
Upper Internals Assembly					
17 x 17 Guide Tube Intermediate Flange	Not Applicable(3)	0.5 Maximum	Static	Possible > 20%(1)	Potentially Susceptible
17 x 17 Guide Tube Lower Flange	RCCA Guide Tube Assemblies (lower flanges)	0.5 Maximum	Static	Possible > 20%(1)	Potentially Susceptible
17 x 17 Guide Tube Lower Guide Plates/Cards	RCCA guide tube assemblies (tubes, housing plates, and guide plates)	0.5 Maximum	Static	Possible > 20%(1)	Potentially Susceptible
Top Mounted Mixing Device, Mixing Device	Upper Internals Assembly (static flow mixers)	0.5 Maximum	Static	Possible > 20%(1)	Potentially Susceptible
Upper Instrumentation Conduit and Supports - Gussets, Clamps, Supports, and Thermocouple Stop	Not Applicable(3)	0.5 Maximum	Static	Possible > 20%(1)	Potentially Susceptible
Upper Support Column - Orifice Base	Upper Internals Assembly (upper support column bases)	0.5 Maximum	Static	≤ 20%(2)	Not Susceptible
Upper Support Column - Mixing Bases	Upper Internals Assembly (static flow mixers)	0.5 Maximum	Static	≤ 20%(2)	Not Susceptible
Lower Internals Assembly					
Lower Support Column – Cap	Lower Internal Assembly (core support, incl'g core support lugs, columns and sleeves)	0.5 Maximum	Static	≤ 20%(2)	Not Susceptible
Core Support Casting	Lower Internal Assembly (core support dome)	0.5 Maximum	Static	≤ 20%(2)	Not Susceptible

Notes:
1. Where component-specific CMTR is not available, the ferrite content is calculated. Allowable variants of the material specification chemistry requirements may result in ferrite content estimations higher than 20%; thus, the ferrite content is identified as potentially exceeding 20%.
2. Conclusion is based on CMTR chemistry data. The CMTRs do not list the element percentage for nitrogen; thus, per the guidance of [47], nitrogen is assumed to be 0.04 percent. The CMTRs do not list an elemental percentage for molybdenum. Therefore, the current Grade CF8 chemistry requirements were reviewed which specify a maximum of 0.5 percent molybdenum. This maximum value is input into Hull's formula.
3. Component was not evaluated within the SGS Unit 2 AMR [1]; however, because it is manufactured with CASS material, it has been evaluated for aging within this AMP.

6.2.8 SE Applicant/Licensee Action Item 8: Submittal of Information for Staff Review and Approval

As addressed in Section 3.5.1 in this SE, applicants/licensees shall make a submittal for NRC review and approval to credit their implementation of MRP-227, as amended by this SE, as an AMP for the RVI components at their facility. This submittal shall include the information identified in Section 3.5.1 of this SE. This is Applicant/Licensee Action Item 8 [6].

SGS Unit 2 Compliance

SGS Unit 2, per the RIS [4], is considered a Category B plant that is expected to submit their RVI AMP/inspection plan based on the guidance of MRP-227-A, consistent with their commitments. Per the LRA [1], SGS Unit 2 has a commitment to submit their RVI inspection plan for approval by the NRC no later than April 18, 2018. The SGS Unit 2 RVI AMP will also be submitted to meet the guidelines provided in the RIS [4].

Conclusion

SGS Unit 2 complies with Applicant/Licensee Action Item 8 of the NRC SE on MRP-227, Revision 0, and therefore meets the requirement for application of MRP-227-A as a strategy for managing age-related material degradation in reactor internals components.

7 PROGRAM ENHANCEMENT AND IMPLEMENTATION SCHEDULE

The requirements of MRP-227-A are based on an 18-month refueling cycle and consider both EFPY and cumulative operation. The information contained in Table 7-1 is based on this information and includes a description of the latest scope of inspections pertaining to the reactor internals AMP. Should a change occur in plant operational practices or should operating experience result in changes to the projections, appropriate updates will be performed on affected plant documentation in accordance with approved procedures.

Table 7-1 SGS Unit 2 Aging Management Program Enhancement and Inspection Implementation Summary

Refueling Outage	Project Month/Year	Estimated EFPY	AMP-Related Scope	Inspection Method and Criteria	Comments
18	Spring 2011	20.1	Not applicable	Not applicable	Not applicable
19	Fall 2012	21.6	Not applicable	Not applicable	Not applicable
20	Spring 2014	23.0	Not applicable	Not applicable	Not applicable
21	Fall 2015	24.5	Not applicable	Not applicable	Not applicable
22	Spring 2017	25.9	Not applicable	Not applicable	Not applicable
23	Fall 2018	27.4	Not applicable	Not applicable	Not applicable
24	Spring 2020	28.8	Not applicable	Not applicable	**Period of Extended Operation begins on April 18, 2020**
25	Fall 2021	30.3	ASME Code Section XI 10-Year ISI.	ASME Code Section XI	Not applicable

Table 7-1 SGS Unit 2 Aging Management Program Enhancement and Inspection Implementation Summary (cont.)

Refueling Outage	Project Month/Year	Estimated EFPY	AMP-Related Scope	Inspection Method and Criteria	Comments
26	Spring 2023	31.7	Initial MRP-227-A augmented inspection for control rod guide tube guide plates (cards), control rod guide tube lower flange welds, upper and lower core barrel flange welds, upper and lower core barrel cylinder girth welds, and thermal shield flexures completed during or before this outage.	MRP-227-A inspections in accordance with MRP-228 specifications	SGS Unit 2 will begin the period of extended operation during Cycle 24. PSEG Nuclear has the option to perform these augmented inspections until 1RF26. The inspection window for these components is within two refueling cycles from the beginning of extended operation. The inspection window for 17x17 standard guide tubes in Westinghouse four-loop plants is 24 to 28 EFPY. As SGS Unit 2 was a participating plant for this analysis, an additional four EFPY can be applied to the initial inspection measurement schedule. Therefore, the initial inspection must be performed before SGS Unit 2 reaches 32 EFPY. Subsequent inspections are based on initial inspection results. See WCAP-17451-P [44] for additional information regarding the inspection schedule and requirements.

Table 7-1 (cont.) SGS Unit 2 Aging Management Program Enhancement and Inspection Implementation Summary

Refueling Outage	Project Month/Year	Estimated EFPY	AMP-Related Scope	Inspection Method and Criteria	Comments
27	Fall 2024	33.2	Initial MRP-227-A augmented inspection for internals hold down spring completed during or before this outage.	MRP-227-A inspections in accordance with MRP-228 specifications	The inspection window for the hold down spring is within three refueling cycles from the beginning of extended operation. If spring height is not sufficient, spring height measurements must be taken during the next two outages.
28	Spring 2026	34.6	Initial MRP-227-A augmented inspection for baffle-former bolts completed during or before this outage.	MRP-227-A inspections in accordance with MRP-228 specifications	The inspection window for baffle-former bolts is between 25 and 35 EFPY. PSEG Nuclear has the option to perform this inspection until RO-28.
29	Fall 2027	36.1	Not applicable	Not applicable	Not applicable
30	Spring 2029	37.5	Not applicable	Not applicable	Not applicable
31	Fall 2030	39.0	Initial MRP-227-A augmented inspections for baffle-former assembly and baffle-edge bolts completed during or before this outage.	MRP-227-A inspections in accordance with MRP-228 specifications	The inspection window for these components is between 20 and 40 EFPY. PSEG Nuclear has the option to perform these inspections until RO-31.
32	Spring 2032	40.4	ASME Code Section XI 10-Year ISI.	ASME Code Section XI	Not applicable
33	Fall 2033	41.9	Subsequent MRP-227-A augmented inspections for control rod guide tube lower flange welds, upper and lower core barrel flange welds, upper and lower core barrel cylinder girth welds, and thermal shield flexures completed on a ten-year interval.	MRP-227-A inspections in accordance with MRP-228 specifications	The inspection window for these components is 10 years after the initial inspection.

Table 7-1 (cont.) SGS Unit 2 Aging Management Program Enhancement and Inspection Implementation Summary

Refueling Outage	Project Month/Year	Estimated EFPY	AMP-Related Scope	Inspection Method and Criteria	Comments
34	Spring 2035	43.3	Subsequent MRP-227-A augmented inspection for baffle-former bolts completed on a ten-year interval.	MRP-227-A inspections in accordance with MRP-228 specifications	The inspection window for these components is 10 years after the initial inspection.
35	Fall 2036	44.8	Not applicable	Not applicable	Not applicable
36	Spring 2038	46.2	Not applicable	Not applicable	Not applicable
37	Fall 2039	47.7	Not applicable	Not applicable	Not applicable
	April 18, 2040	48	Not applicable	Not applicable	Renewed Operating License expires April 18, 2040

8 IMPLEMENTING DOCUMENTS

The SGS Unit 2 PWR Vessel Internals AMP is implemented through PSEG procedure ER-AP-333, "Pressurized Water Reactor Internals Aging Management Program", Rev. 0 [2]. It also credits existing aging management programs. The SGS Unit 2 RVI AMP references the Water Chemistry Program and the ASME Code Section XI ISI, Subsections IWB, IWC, and IWD Program. MRP-227-A augmented examinations (Appendix C), recommended as a result of industry programs, will be included in the existing ASME Section XI Program.

PSEG documents associated with the existing SGS Programs and considered to be implementing documents of the SGS Unit 2 PWR Vessel Internals Program are:

- CY-AP-120-1000, Primary Strategic Water Chemistry Plan for Recirculating Steam Generator Plants [19]
- CY-AP-120-100, Reactor Coolant System Chemistry [20]
- ER-AA-330, Conduct of Inservice Inspection Activities [10]
- SA-PBD-AMP-X.M1, Metal Fatigue of Reactor Coolant Pressure Boundary [40]

The SGS Unit 2 RVI AMP relies on the Water Chemistry Program for maintaining high water purity to reduce susceptibility to cracking due to SCC. The Water Chemistry Program was evaluated and found to be consistent with GALL Section XI.M2, Water Chemistry [18]. Additional procedures may be updated or created as OE for augmented examinations is accumulated.

Based on this information, the updated AMP for SGS Unit 2 RVI provides reasonable assurance that the aging effects will be managed such that the components within the scope of license renewal will continue to perform their intended functions consistent with the CLB for the period of extended operation.

9 REFERENCES

1. "License Renewal Application, Salem Nuclear Generating Station, Unit 1 Facility Operation License No. DPR-70, Unit 2 Facility Operating License No. DPR-75."
2. SGS Program ER-AP-333, Revision 0, "Pressurized Water Reactor Internals Aging Management Program."
3. U.S. Nuclear Regulatory Commission Document, NUREG-2101, "Safety Evaluation Report Related to the License Renewal of Salem Nuclear Generating Station, Docket Numbers 50-272 and 50-311," June 2011.
4. U.S. Nuclear Regulatory Commission Document, "NRC Regulatory Issue Summary 2011-07 License Renewal Submittal Information for Pressurized Water Reactor Internals Aging Management," July 21, 2011 (NRC ADAMS Accession No. ML111990086).
5. ASME Boiler and Pressure Vessel Code Section XI, 2004 Edition, no Addenda.
6. *Materials Reliability Program: Pressurized Water Reactor Internals Inspection and Evaluation Guidelines (MRP-227-A).* EPRI, Palo Alto, CA: 2011. 1022863.
7. U.S. Nuclear Regulatory Commission, Code of Federal Regulations, 10 CFR Part 54, "Requirements for Renewal of Operating Licenses for Nuclear Power Plants," August 2007.
8. SGS Units 1 and 2, Program Basis Document SA-PBD-AMP-XI.M16, Revision 3, "GALL Program XI.M16 - PWR Vessel Internals," January 14, 2011.
9. PSEG Report, Revision 27, "Salem Nuclear Generating Station Unit 1 and Unit 2 Updated Final Safety Analysis Report," November 25, 2013.
10. SGS Program ER-AA-330, Revision 10, Conduct of Inservice Inspection Activities," December 2, 2011.
11. U.S. Nuclear Regulatory Commission Document, NUREG-1800, "U.S. NRC Standard Review Plan for the Review of License Renewal Applications for Nuclear Power Plants (SRP-LR)," July 2001.
12. Westinghouse Report, WCAP-14577, Revision 1-A, "License Renewal Evaluation: Aging Management for Reactor Internals," March 2001.
13. *Pressurized Water Reactor Primary Water Chemistry Guidelines, Revision 6,* EPRI, Palo Alto, CA; 2007, 1014986.
14. *Materials Reliability Program: Screening, Categorization and Ranking of Reactor Internals Components of Westinghouse and Combustion Engineering PWR Design (MRP-191).* EPRI, Palo Alto, CA: 2006. 1013234.
15. *Materials Reliability Program: Inspection Standard for PWR Internals – 2012 Update (MRP-228, Rev 1).* EPRI, Palo Alto, CA: 2012. 1025147.
16. Westinghouse Report, WCAP-17096-NP, Revision 2, "Reactor Internals Acceptance Criteria Methodology and Data Requirements," December 2009.
17. NEI 03-08, Revision 2, "Guidelines for the Management of Materials Issues," Nuclear Energy Institute, Washington, DC, January 2010.

18. U.S. Nuclear Regulatory Commission Document, NUREG-1801, Revision 2, "Generic Aging Lessons Learned (GALL) Report," December 2010.
19. SGS Document CY-AP-120-1000, Revision 8, "Primary Strategic Water Chemistry Plan for Recirculating Steam Generator Plants," June 10, 2013.
20. SGS Document CY-AP-120-100, Revision 16, "Reactor Coolant System Chemistry," May 9, 2013.
21. U.S. Nuclear Regulatory Commission Bulletin 88-09, "Thimble Tube Thinning in Westinghouse Reactors," July 26, 1988.
22. Westinghouse Report, WCAP-15028, Revision 0, "Guide Tube Cold-Worked 316 Replacement Support Pin Development Program," March 1998 (Westinghouse Proprietary Class 2).
23. SGS Quality Procedure LS-AA-125, Revision 17, "Corrective Action Program (CAP) Procedure," November 1, 2013.
24. SGS Quality Procedure LS-AA-125-1001, Revision 9, "Root Cause Evaluation Manual," May 6, 2013.
25. SGS Quality Procedure LS-AA-125-1002, Revision 7, "Common Cause Analysis Manual," February 8, 2011.
26. SGS Quality Procedure LS-AA-125-1003, Revision 13, "Apparent Cause Evaluation Manual," September 5, 2013.
27. SGS Quality Procedure LS-AA-125-1004, Revision 4, "Effectiveness Review Manual," June 23, 2011.
28. SGS Quality Procedure LS-AA-125-1005, Revision 6, "Coding and Analysis Model," December 18, 2007.
29. SGS Quality Procedure LA-AA-125-1006, Revision 5, "Department and Station Roll-Up Meetings (DRUM SRUM)," August 1, 2013.
30. SGS Document OP-AA-108-111, Revision 7, "Adverse Condition Monitoring and Contingency Planning," September 20, 2012.
31. SGS Document ER-SH-330-0009, Revision 5, "Repair Program Manual," October 12, 2012.
32. Westinghouse Report, WCAP-15270, Revision 0, "Determination of Acceptable Baffle-Barrel-Bolting for Four-Loop Westinghouse Downflow and Converted Upflow Domestic Plants," November 2001 (Westinghouse Proprietary Class 2).
33. SGS Document HU-AA-104-101, Revision 5, "Procedure Use and Adherence," May 29, 2012.
34. SGS Document AD-AA-101, Revision 20, "Processing of Procedures and Training and Reference Materials (T&RMs)," January 6, 2014.
35. U.S. Nuclear Regulatory Commission Information Notice 84-18, "Stress Corrosion Cracking in Pressurized Water Reactor Systems," March 7, 1984.
36. U.S. Nuclear Regulatory Commission Information Notice 98-11, "Cracking of Reactor Vessel Internal Baffle Former Bolts in Foreign Plants," March 25, 1998.
37. SGS Document LS-AA-115, Revision 13, "Operating Experience Procedure," March 16, 2012.
38. Salem Unit 2 Split Pin Replacement, NUCP 80008845, November 11, 2001.

39. SGS Document SA-PBD-AMP-XI.M37, Revision 3, "Flux Thimble Tube Inspection," February 1, 2011.

40. SGS Document SA-PBD-AMP-X.M1, Revision 3, "Metal Fatigue of Reactor Coolant Pressure Boundary," January 14, 2011.

41. SGS Transmittal of Design Information, S-TODI-2013-0003, Revision 1, "Westinghouse Request for Information in Support of PSEG for PA-MSC-0983," December 12, 2013.

42. U.S. Nuclear Regulatory Commission Document, NUREG-1801, Revision 1, "Generic Aging Lessons Learned (GALL) Report," September 2005.

43. U.S. Nuclear Regulatory Commission Letter, "License Renewal Issue No. 98-0030, *Thermal Aging Embrittlement of Cast Austenitic Stainless Steel Components*," May 19, 2000 (NRC ADAMS Accession No. ML003717179).

44. Westinghouse Report, WCAP-17451-P, Revision 1, "Reactor Internals Guide Tube Wear – Westinghouse Domestic Fleet Operational Projections," October 2013 (Westinghouse Proprietary Class 2).

45. Westinghouse Letter, LTR-RIAM-13-136, Revision 0, "Summary of Salem Units 1 and 2 Expert Elicitation Panel Meeting Minutes for Reactor Internals Components Identified as Potentially Cast Austenitic Stainless Steel," January 28, 2014 (Westinghouse Proprietary Class 2).

46. Westinghouse Letter, LTR-RIDA-14-14, Revision 1, "Salem Units 1 and 2 - Lower Internals Hold-down Spring Height Criteria," June 26, 2014 (Westinghouse Proprietary Class 2).

47. U.S. Nuclear Regulatory Commission Report, NUREG/CR-4513, Revision 1, "Estimation of Fracture Toughness of Cast Stainless Steels During Thermal Aging in LWR Systems," May 1994 (NRC ADAMS Accession No. ML052360554).

48. EPRI Letter, MRP 2014-006, "*Materials Reliability Program:* Pressurized Water Reactor Internals Inspection and Evaluation Guidelines (MRP-227-A), EPRI, Palo Alto, CA: 2011. 1022863, Transmittal of Interim Guidance," February 18, 2014.

APPENDIX A
ILLUSTRATIONS

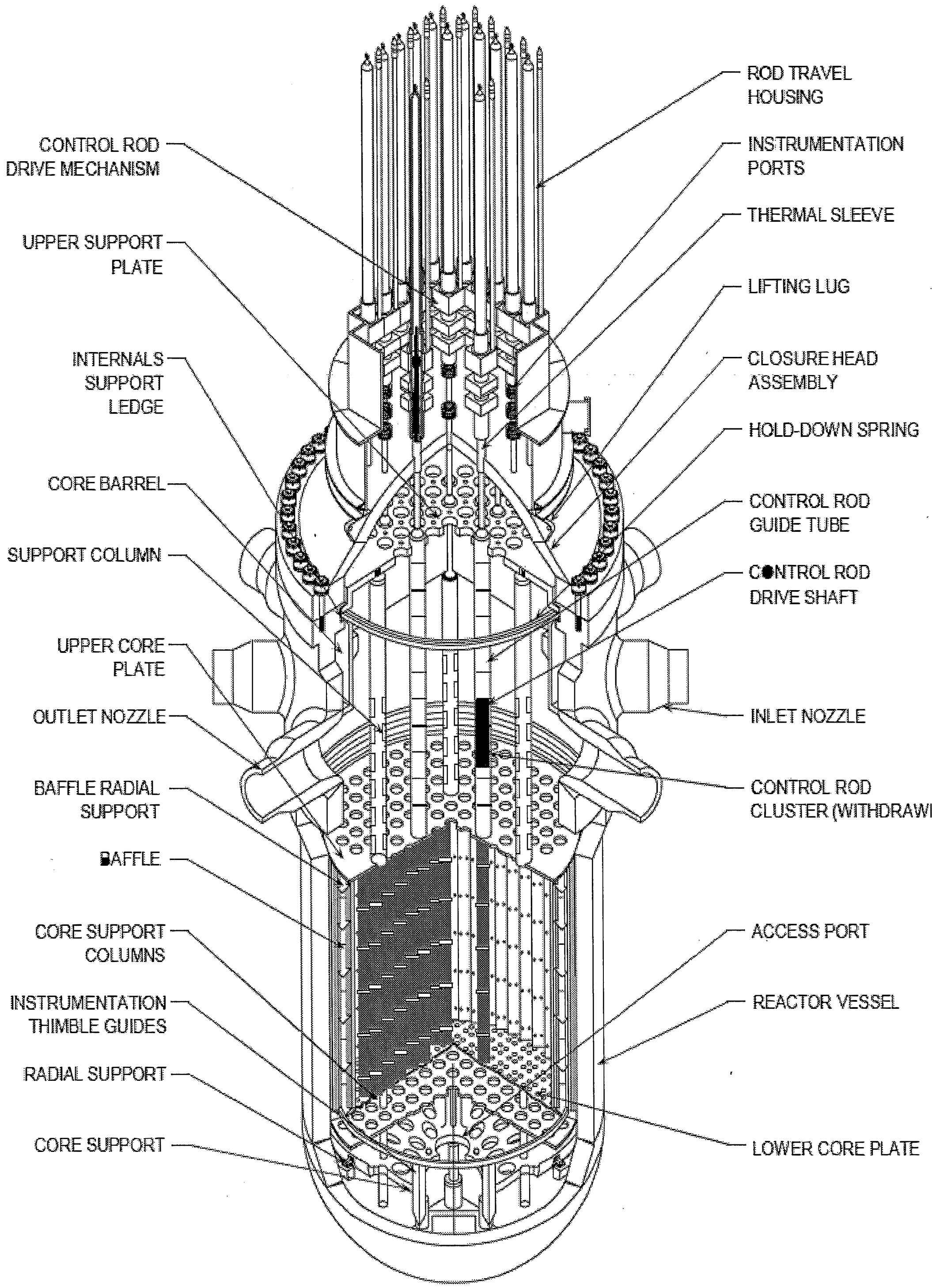

Figure A-1 Illustration of Typical Westinghouse Internals

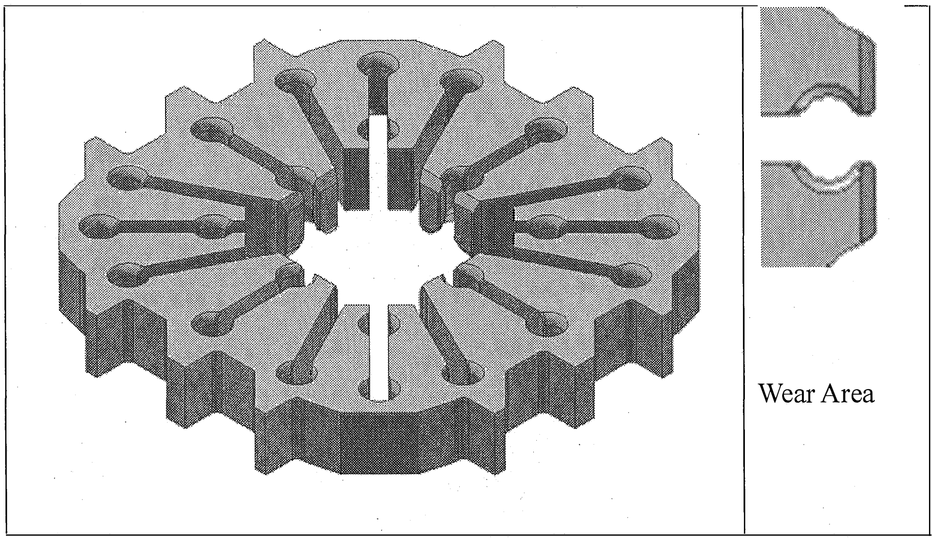

Figure A-2 Typical Westinghouse Control Rod Guide Card

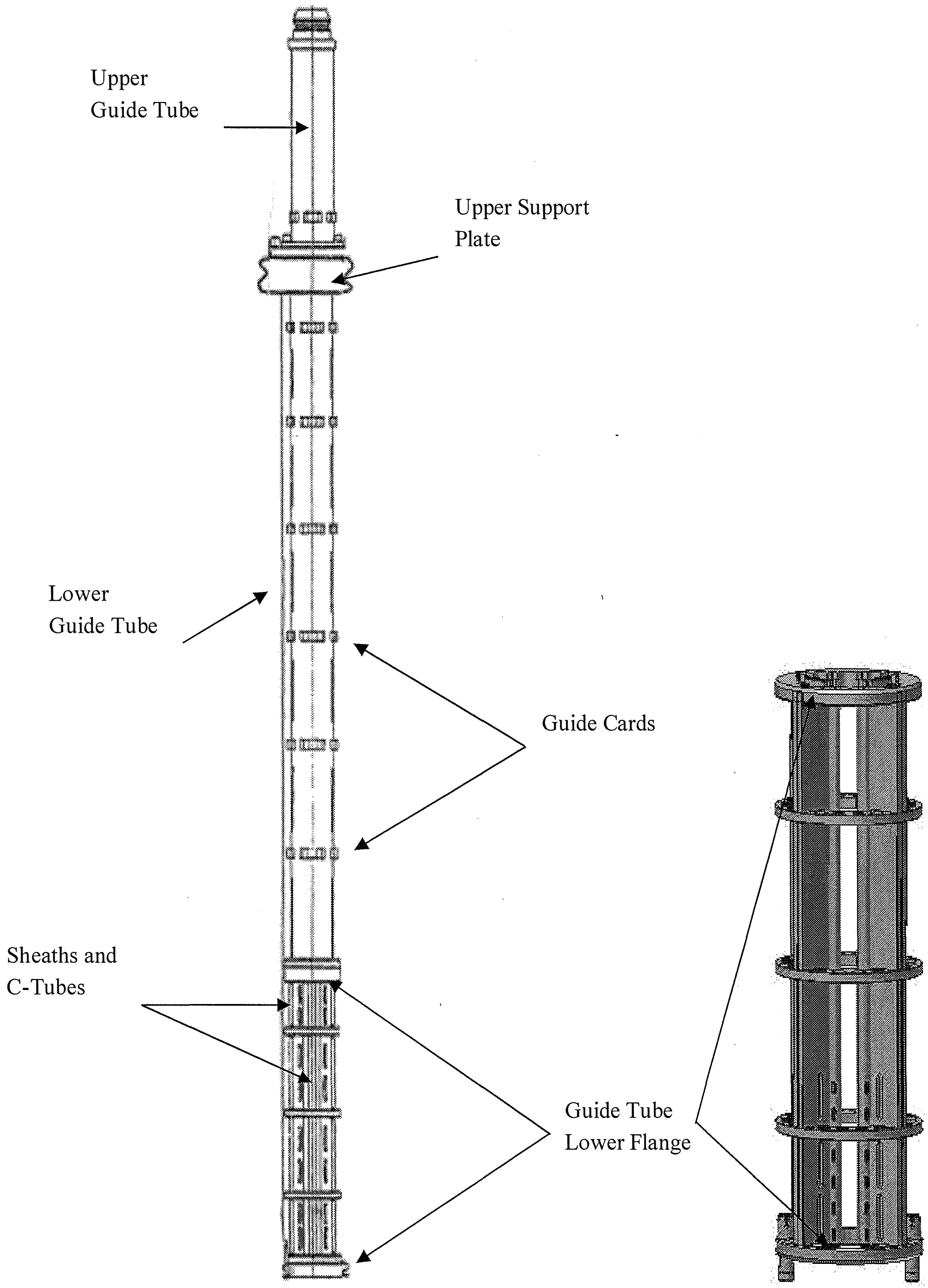

Figure A-3 Lower Section of Control Rod Guide Tube Assembly

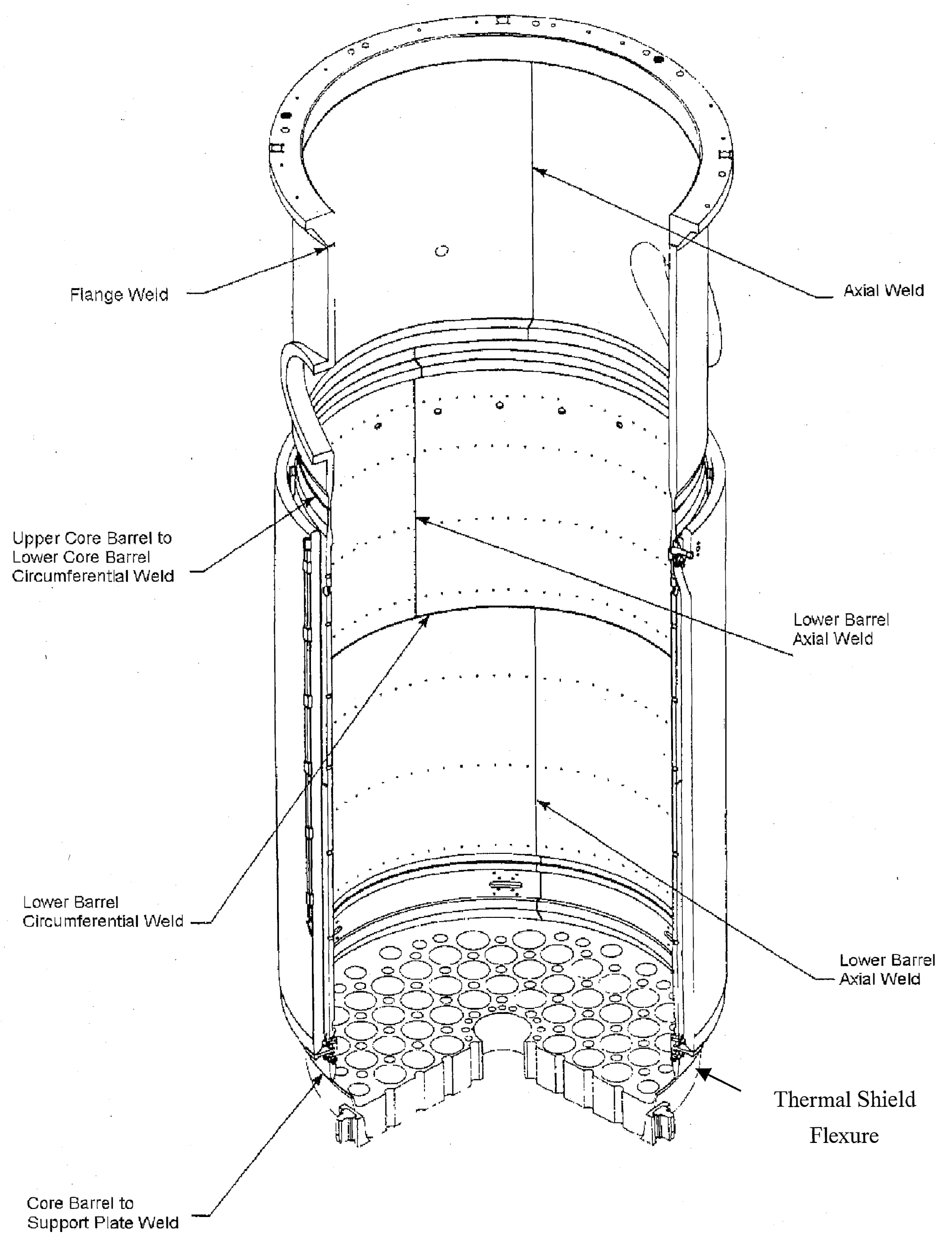

Figure A-4 Major Core Barrel Welds

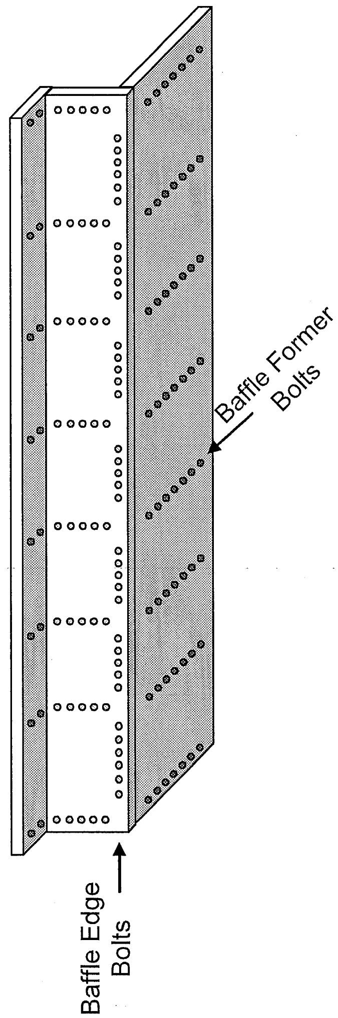

Figure A-5 Bolting Systems used in Typical Westinghouse Core Baffles
(Baffle-edge bolts are applicable for SGS Unit 2)

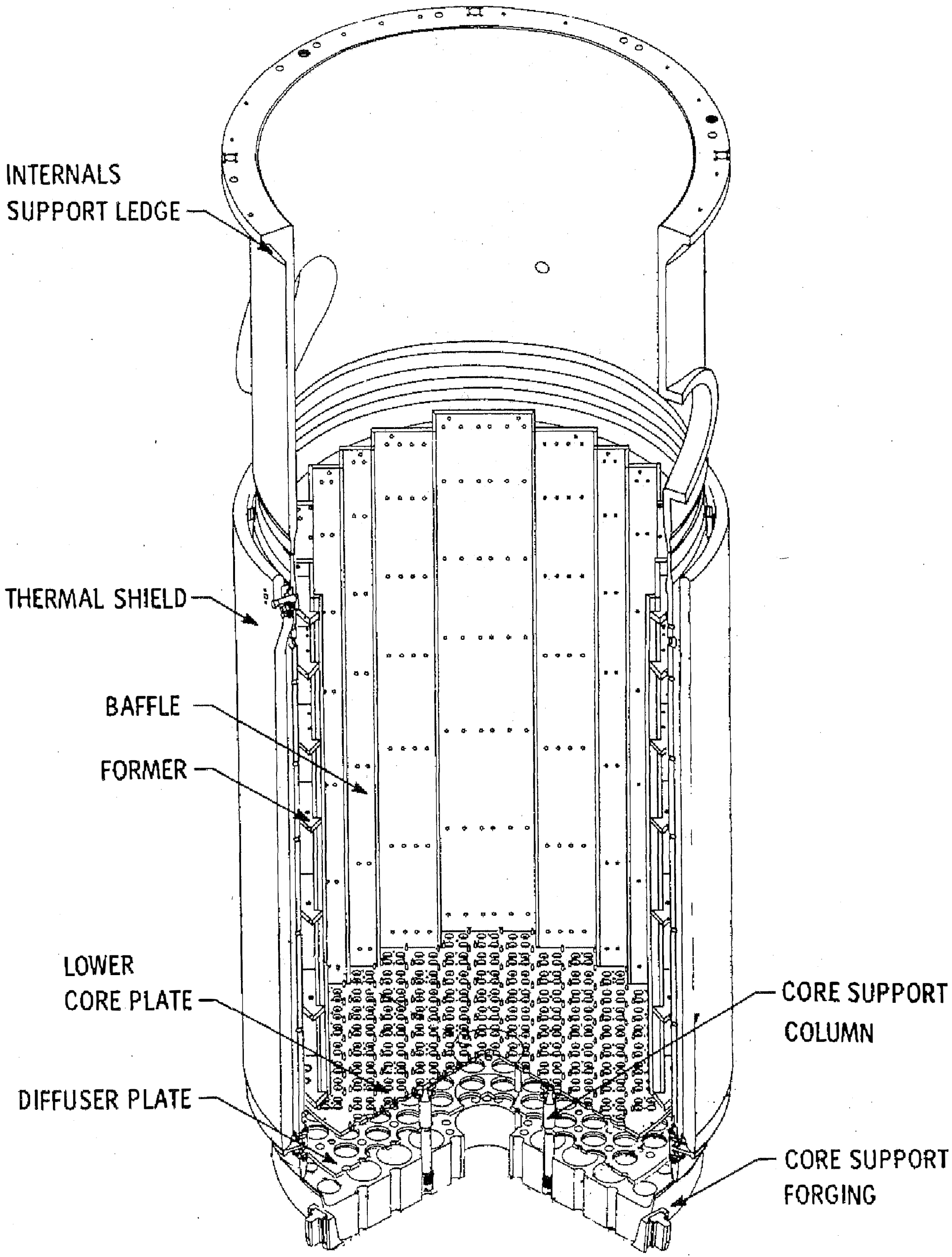

Figure A-6 Core Baffle/Barrel Structure

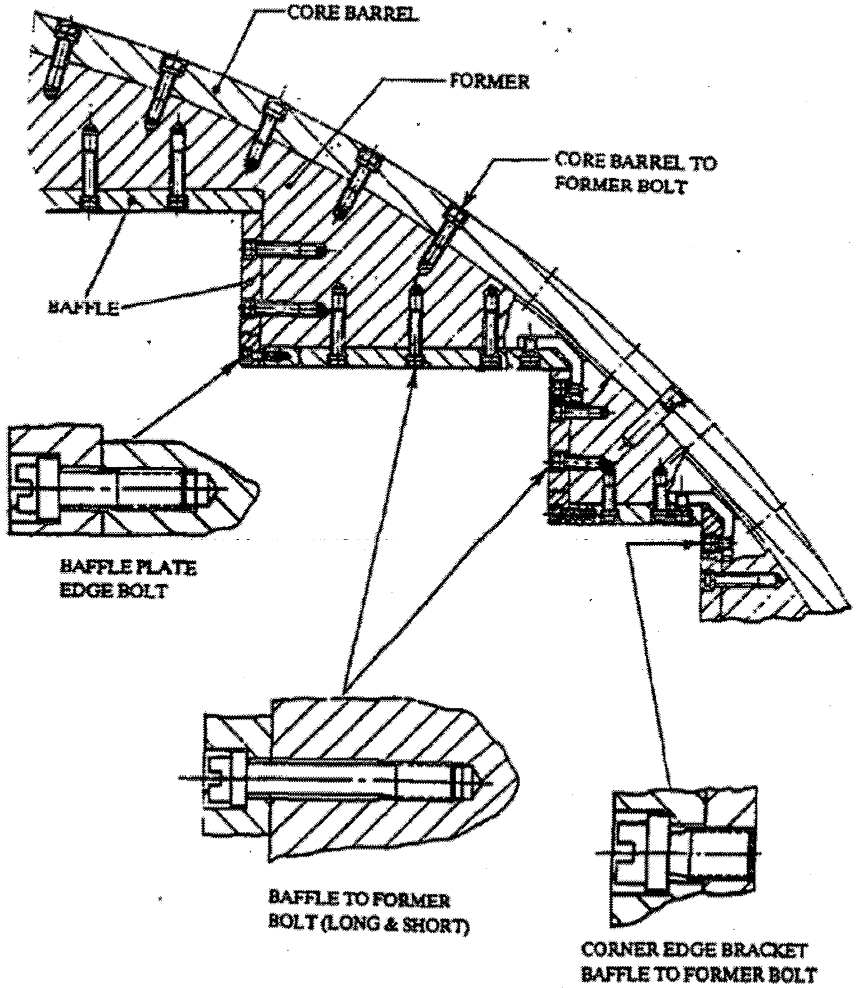

Figure A-7 Bolting in a Typical Westinghouse Baffle/Former Structure
(Baffle-edge bolts are applicable for SGS Unit 2)

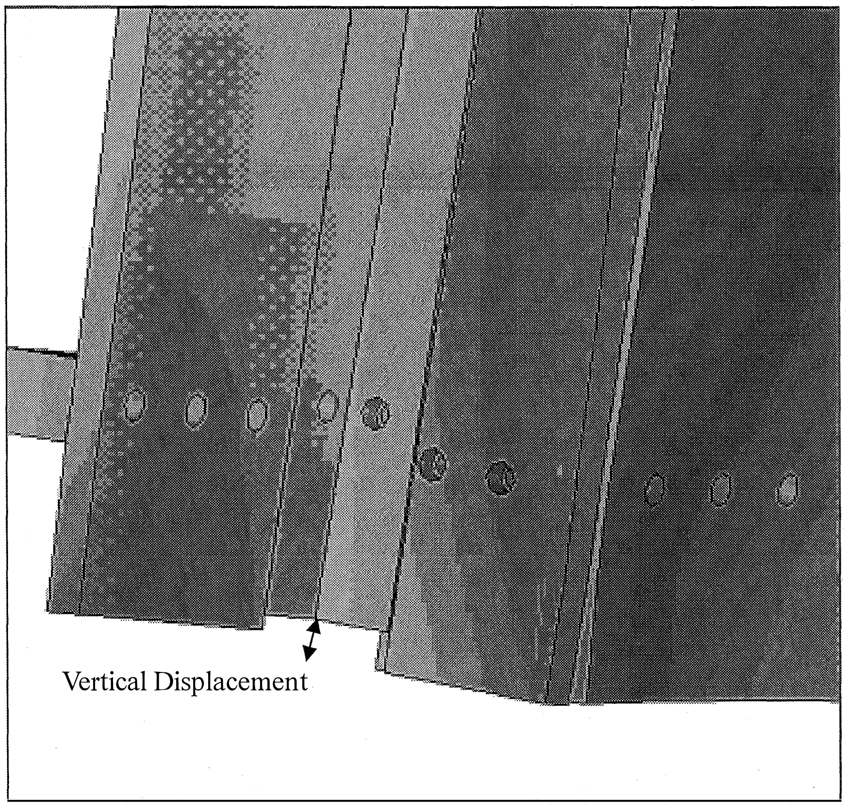

Figure A-8 Example of the Vertical Displacement between the Baffle Plates and Bracket at the Bottom of the Baffle-Former-Barrel Assembly which could Result from Void Swelling

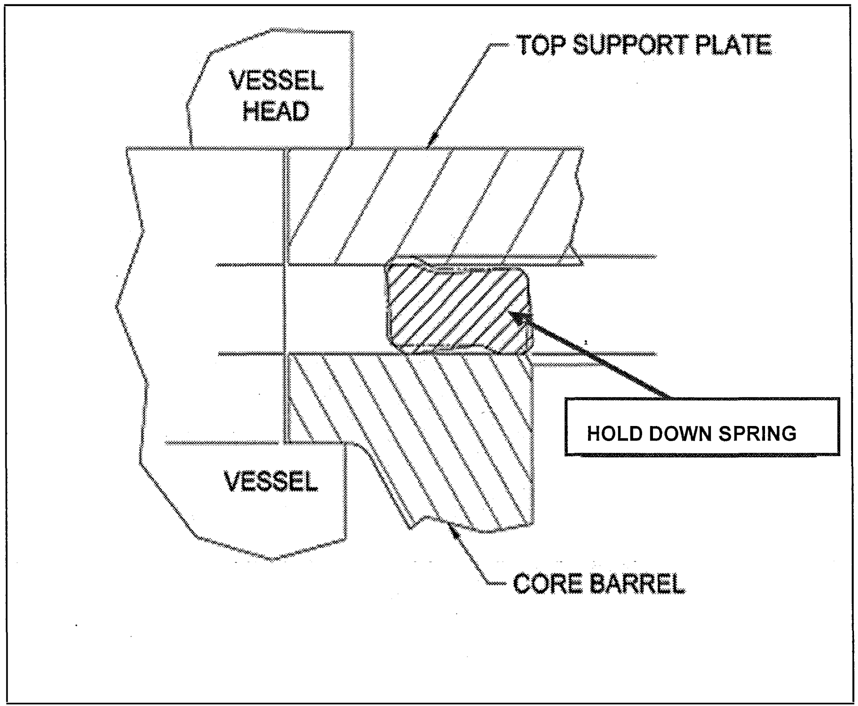

Figure A-9 Schematic Cross-Sections of the Westinghouse Hold Down Spring

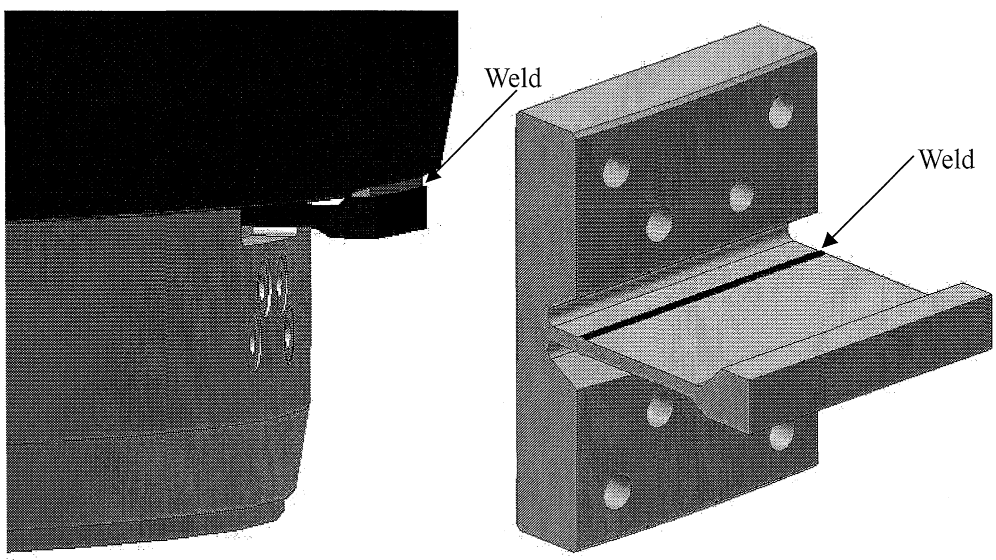

Figure A-10 Typical Thermal Shield Flexure

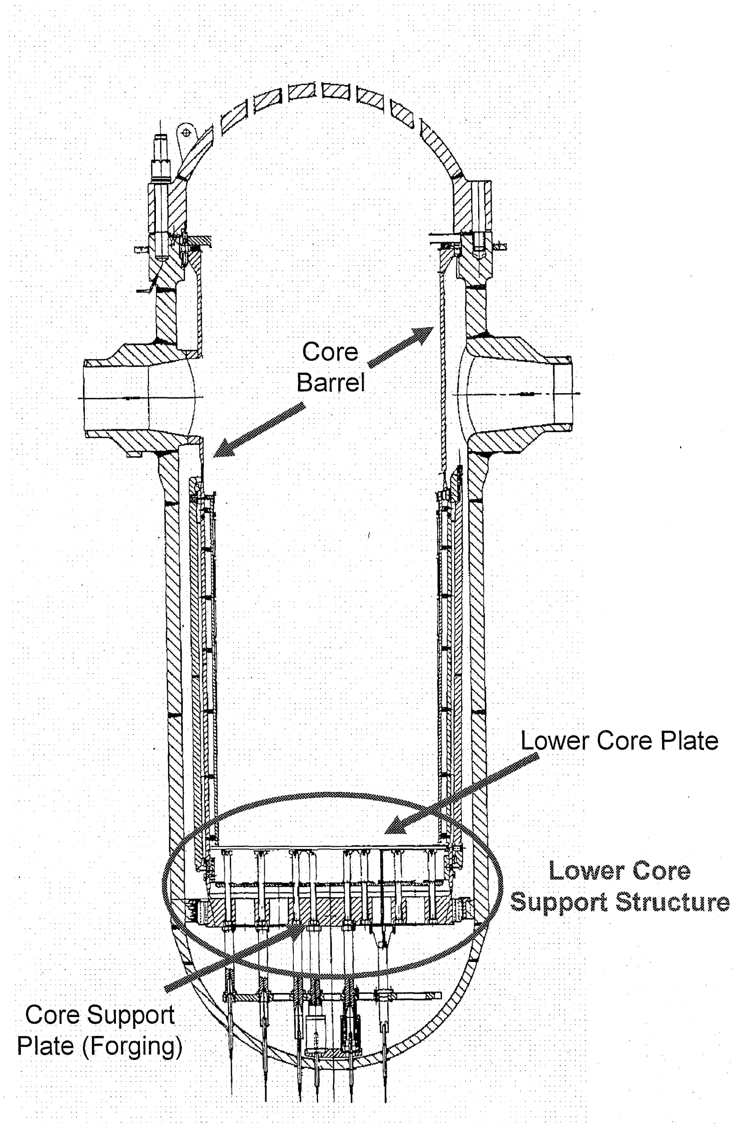

Figure A-11 Lower Core Support Structure

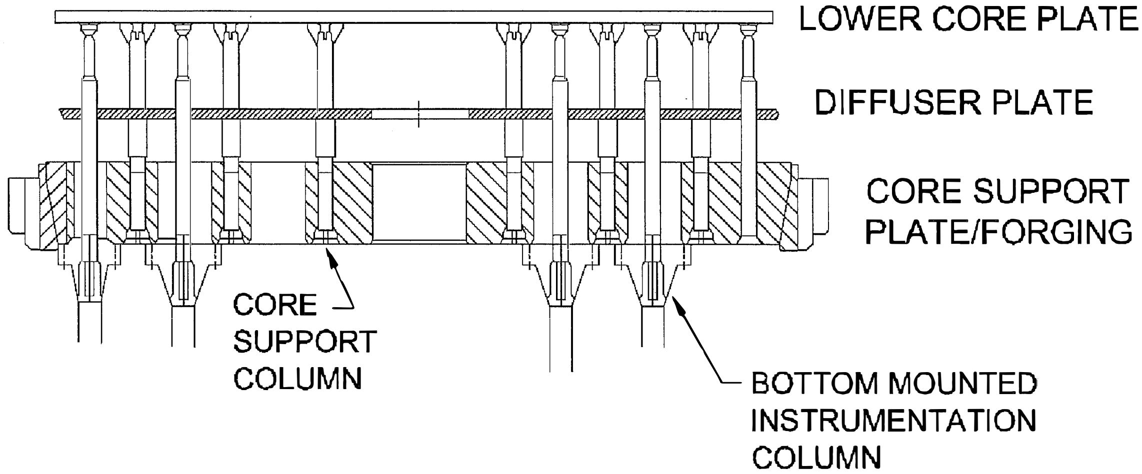

Figure A-12 Lower Core Support Structure – Cross-Section

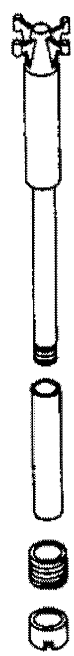

Figure A-13 Typical Core Support Column

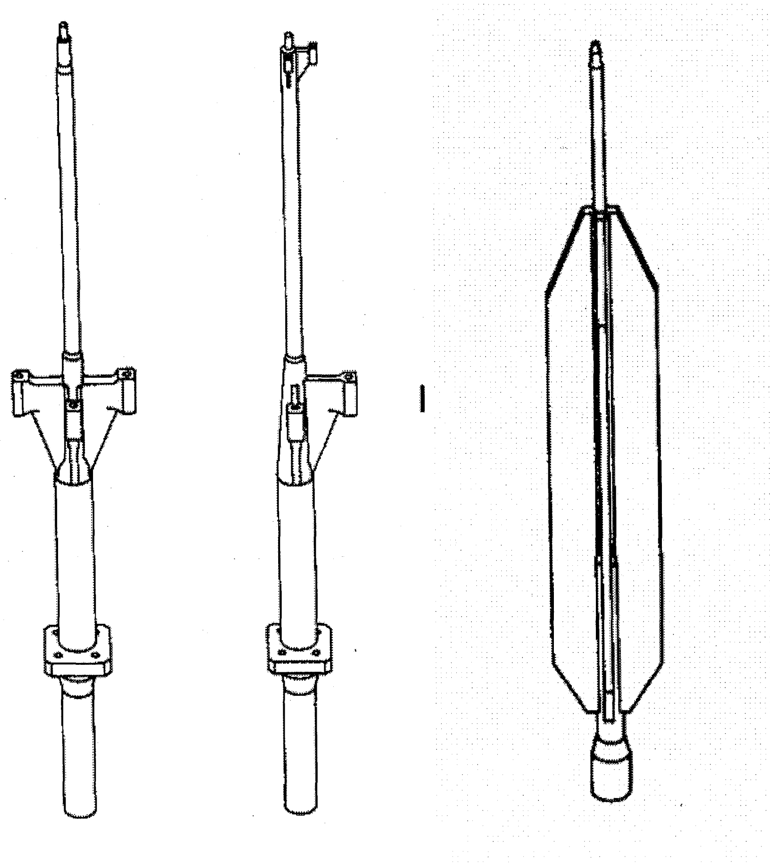

Figure A-14 Examples of BMI Column Designs

APPENDIX B
SALEM NUCLEAR GENERATING STATION LICENSE RENEWAL AGING MANAGEMENT REVIEW SUMMARY TABLES

The content in Table B-1 is extracted from Table 3.1.2-3 "Reactor Vessel Internals Summary of Aging Management Evaluation" of the SGS LRA [1]. Only those items applicable to RVI (according to the LRA) were imported into Table B-1 from the LRA. Per reference [41], it was identified that some components listed in this table were not consistent with the RVI design for SGS Unit 2. Notes have been added in the comments column to identify these components within Table B-1.

Table B-1 Summary of SGS LRA Table 3.1.2-3 – Reactor Vessel Internals Summary of Aging Management Evaluation

Component Type	Material	Aging Effect Requiring Management	Aging Management Programs	Comments
Control Rod Assembly	Not Applicable	Not Applicable	Not Applicable	Not Applicable
Core Barrel (lower)	Stainless Steel	Changes in Dimensions/Void Swelling	PWR Vessel Internals	Not Applicable
	Stainless Steel	Cracking/Stress Corrosion Cracking, Irradiation-Assisted Stress Corrosion Cracking	PWR Vessel Internals	Not Applicable
	Stainless Steel	Cracking/Stress Corrosion Cracking, Irradiation-Assisted Stress Corrosion Cracking	Water Chemistry	Not Applicable
	Stainless Steel	Loss of Fracture Toughness/Neutron Irradiation Embrittlement, Void Swelling	PWR Vessel Internals	Not Applicable
	Stainless Steel	Loss of Material/Pitting and Crevice Corrosion	Water Chemistry	Not Applicable

Table B-1 Summary of SGS LRA Table 3.1.2-3 – Reactor Vessel Internals Summary of Aging Management Evaluation

(cont.)

Component Type	Material	Aging Effect Requiring Management	Aging Management Programs	Comments
Core Barrel (upper)	Stainless Steel	Changes in Dimensions/Void Swelling	PWR Vessel Internals	Not Applicable
	Stainless Steel	Cracking/Stress Corrosion Cracking, Irradiation-Assisted Stress Corrosion Cracking	PWR Vessel Internals	Not Applicable
	Stainless Steel	Cracking/Stress Corrosion Cracking, Irradiation-Assisted Stress Corrosion Cracking	Water Chemistry	Not Applicable
	Stainless Steel	Loss of Fracture Toughness/Neutron Irradiation Embrittlement, Void Swelling	PWR Vessel Internals	Not Applicable
	Stainless Steel	Loss of Material/Pitting and Crevice Corrosion	Water Chemistry	Not Applicable
Core Barrel Assembly (alignment pins)	Stainless Steel	Changes in Dimensions/Void Swelling	PWR Vessel Internals	Not Applicable
	Stainless Steel	Cracking/Stress Corrosion Cracking, Irradiation-Assisted Stress Corrosion Cracking	PWR Vessel Internals	Not Applicable
	Stainless Steel	Cracking/Stress Corrosion Cracking, Irradiation-Assisted Stress Corrosion Cracking	Water Chemistry	Not Applicable
	Stainless Steel	Loss of Fracture Toughness/Neutron Irradiation Embrittlement, Void Swelling	PWR Vessel Internals	Not Applicable
	Stainless Steel	Loss of Material/Pitting and Crevice Corrosion	Water Chemistry	Not Applicable
	Stainless Steel	Loss of Preload/Stress Relaxation	PWR Vessel Internals	Not Applicable

Table B-1 Summary of SGS LRA Table 3.1.2-3 – Reactor Vessel Internals Summary of Aging Management Evaluation

(cont.)

Component Type	Material	Aging Effect Requiring Management	Aging Management Programs	Comments
Core Barrel Assembly (baffle bolt lock bars)	Stainless Steel	Changes in Dimensions/Void Swelling	PWR Vessel Internals	Not Applicable
	Stainless Steel	Cracking/Stress Corrosion Cracking, Irradiation-Assisted Stress Corrosion Cracking	PWR Vessel Internals	Not Applicable
	Stainless Steel	Cracking/Stress Corrosion Cracking, Irradiation-Assisted Stress Corrosion Cracking	Water Chemistry	Not Applicable
	Stainless Steel	Loss of Fracture Toughness/Neutron Irradiation Embrittlement, Void Swelling	PWR Vessel Internals	Not Applicable
	Stainless Steel	Loss of Material/Pitting and Crevice Corrosion	Water Chemistry	Not Applicable
	Stainless Steel	Loss of Preload/Stress Relaxation	PWR Vessel Internals	Not Applicable
Core Barrel Assembly (baffle former assembly - plates)	Stainless Steel	Changes in Dimensions/Void Swelling	PWR Vessel Internals	Not Applicable
	Stainless Steel	Cracking/Stress Corrosion Cracking, Irradiation-Assisted Stress Corrosion Cracking	PWR Vessel Internals	Not Applicable
	Stainless Steel	Cracking/Stress Corrosion Cracking, Irradiation-Assisted Stress Corrosion Cracking	Water Chemistry	Not Applicable
	Stainless Steel	Loss of Fracture Toughness/Neutron Irradiation Embrittlement, Void Swelling	PWR Vessel Internals	Not Applicable
	Stainless Steel	Loss of Material/Pitting and Crevice Corrosion	Water Chemistry	Not Applicable

Table B-1 Summary of SGS LRA Table 3.1.2-3 – Reactor Vessel Internals Summary of Aging Management Evaluation

(cont.)

Component Type	Material	Aging Effect Requiring Management	Aging Management Programs	Comments
Core Barrel Assembly (core barrel to thermal shield bolts and dowels)	Stainless Steel Bolting	Changes in Dimensions/Void Swelling	PWR Vessel Internals	Not Applicable
	Stainless Steel Bolting	Cracking/Stress Corrosion Cracking, Irradiation-Assisted Stress Corrosion Cracking	PWR Vessel Internals	Not Applicable
	Stainless Steel Bolting	Cracking/Stress Corrosion Cracking, Irradiation-Assisted Stress Corrosion Cracking	Water Chemistry	Not Applicable
	Stainless Steel Bolting	Loss of Fracture Toughness/Neutron Irradiation Embrittlement, Void Swelling	PWR Vessel Internals	Not Applicable
	Stainless Steel Bolting	Loss of Material/Pitting and Crevice Corrosion	Water Chemistry	Not Applicable
	Stainless Steel Bolting	Loss of Preload/Stress Relaxation	PWR Vessel Internals	Not Applicable
Core Barrel Assembly (flange)	Stainless Steel	Changes in Dimensions/Void Swelling	PWR Vessel Internals	Not Applicable
	Stainless Steel	Cracking/Stress Corrosion Cracking, Irradiation-Assisted Stress Corrosion Cracking	PWR Vessel Internals	Not Applicable
	Stainless Steel	Cracking/Stress Corrosion Cracking, Irradiation-Assisted Stress Corrosion Cracking	Water Chemistry	Not Applicable
	Stainless Steel	Loss of Fracture Toughness/Neutron Irradiation Embrittlement, Void Swelling	PWR Vessel Internals	Not Applicable
	Stainless Steel	Loss of Material/Pitting and Crevice Corrosion	Water Chemistry	Not Applicable

Table B-1 Summary of SGS LRA Table 3.1.2-3 – Reactor Vessel Internals Summary of Aging Management Evaluation

(cont.)

Component Type	Material	Aging Effect Requiring Management	Aging Management Programs	Comments
Core Barrel Assembly (lock bar, baffle-former, barrel-former, and baffle-edge bolting)	Stainless Steel Bolting	Changes in Dimensions/Void Swelling	PWR Vessel Internals	Not Applicable
	Stainless Steel Bolting	Cracking/Stress Corrosion Cracking, Irradiation-Assisted Stress Corrosion Cracking	PWR Vessel Internals	Not Applicable
	Stainless Steel Bolting	Cracking/Stress Corrosion Cracking, Irradiation-Assisted Stress Corrosion Cracking	Water Chemistry	Not Applicable
	Stainless Steel Bolting	Loss of Fracture Toughness/Neutron Irradiation Embrittlement, Void Swelling	PWR Vessel Internals	Not Applicable
	Stainless Steel Bolting	Loss of Material/Pitting and Crevice Corrosion	Water Chemistry	Not Applicable
	Stainless Steel Bolting	Loss of Preload/Stress Relaxation	PWR Vessel Internals	Not Applicable
Core Barrel Assembly (outlet nozzle)	Stainless Steel	Changes in Dimensions/Void Swelling	PWR Vessel Internals	Not Applicable
	Stainless Steel	Cracking/Stress Corrosion Cracking, Irradiation-Assisted Stress Corrosion Cracking	PWR Vessel Internals	Not Applicable
	Stainless Steel	Cracking/Stress Corrosion Cracking, Irradiation-Assisted Stress Corrosion Cracking	Water Chemistry	Not Applicable
	Stainless Steel	Loss of Fracture Toughness/Neutron Irradiation Embrittlement, Void Swelling	PWR Vessel Internals	Not Applicable
	Stainless Steel	Loss of Material/Pitting and Crevice Corrosion	Water Chemistry	Not Applicable

Table B-1 Summary of SGS LRA Table 3.1.2-3 – Reactor Vessel Internals Summary of Aging Management Evaluation

(cont.)

Component Type	Material	Aging Effect Requiring Management	Aging Management Programs	Comments
Core Barrel Assembly (spray nozzles)	Stainless Steel	Changes in Dimensions/Void Swelling	PWR Vessel Internals	Not Applicable
	Stainless Steel	Cracking/Stress Corrosion Cracking, Irradiation-Assisted Stress Corrosion Cracking	PWR Vessel Internals	Not Applicable
	Stainless Steel	Cracking/Stress Corrosion Cracking, Irradiation-Assisted Stress Corrosion Cracking	Water Chemistry	Not Applicable
	Stainless Steel	Loss of Fracture Toughness/Neutron Irradiation Embrittlement, Void Swelling	PWR Vessel Internals	Not Applicable
	Stainless Steel	Loss of Material/Pitting and Crevice Corrosion	Water Chemistry	Not Applicable
Fuel Assembly (Short-lived)	Not Applicable	Not Applicable	Not Applicable	Not Applicable
Lower Internal Assembly (axial flexures: thermal shield to core barrel)	Stainless Steel	Changes in Dimensions/Void Swelling	PWR Vessel Internals	Not Applicable
	Stainless Steel	Cracking/Stress Corrosion Cracking, Irradiation-Assisted Stress Corrosion Cracking	PWR Vessel Internals	Not Applicable
	Stainless Steel	Cracking/Stress Corrosion Cracking, Irradiation-Assisted Stress Corrosion Cracking	Water Chemistry	Not Applicable
	Stainless Steel	Loss of Fracture Toughness/Neutron Irradiation Embrittlement, Void Swelling	PWR Vessel Internals	Not Applicable
	Stainless Steel	Loss of Material/Pitting and Crevice Corrosion	Water Chemistry	Not Applicable
	Stainless Steel	Loss of Material/Wear	ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD	Not Applicable

Table B-1 Summary of SGS LRA Table 3.1.2-3 – Reactor Vessel Internals Summary of Aging Management Evaluation

(cont.)

Component Type	Material	Aging Effect Requiring Management	Aging Management Programs	Comments
Lower Internal Assembly (clevis block bolts)	Nickel Alloy	Changes in Dimensions/Void Swelling	PWR Vessel Internals	Not Applicable
	Nickel Alloy	Cracking/Stress Corrosion Cracking, Irradiation-Assisted Stress Corrosion Cracking	PWR Vessel Internals	Not Applicable
	Nickel Alloy	Cracking/Stress Corrosion Cracking, Irradiation-Assisted Stress Corrosion Cracking	Water Chemistry	Not Applicable
	Nickel Alloy	Loss of Fracture Toughness/Neutron Irradiation Embrittlement, Void Swelling	PWR Vessel Internals	Not Applicable
	Nickel Alloy	Loss of Material/Pitting and Crevice Corrosion	Water Chemistry	Not Applicable
	Nickel Alloy	Loss of Preload/Stress Relaxation	PWR Vessel Internals	Not Applicable
Lower Internal Assembly (clevis block lock keys)	Nickel Alloy	Changes in Dimensions/Void Swelling	PWR Vessel Internals	Not Applicable
	Nickel Alloy	Cracking/Stress Corrosion Cracking, Irradiation-Assisted Stress Corrosion Cracking	PWR Vessel Internals	Not Applicable
	Nickel Alloy	Cracking/Stress Corrosion Cracking, Irradiation-Assisted Stress Corrosion Cracking	Water Chemistry	Not Applicable
	Nickel Alloy	Loss of Material/Pitting and Crevice Corrosion	Water Chemistry	Not Applicable
	Nickel Alloy	Loss of Material/Wear	ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD	Not Applicable

Table B-1 Summary of SGS LRA Table 3.1.2-3 – Reactor Vessel Internals Summary of Aging Management Evaluation

(cont.)

Component Type	Material	Aging Effect Requiring Management	Aging Management Programs	Comments
Lower Internal Assembly (clevis blocks and inserts)	Nickel Alloy	Changes in Dimensions/Void Swelling	PWR Vessel Internals	Not Applicable
	Nickel Alloy	Cracking/Stress Corrosion Cracking, Irradiation-Assisted Stress Corrosion Cracking	PWR Vessel Internals	Not Applicable
	Nickel Alloy	Cracking/Stress Corrosion Cracking, Irradiation-Assisted Stress Corrosion Cracking	Water Chemistry	Not Applicable
	Nickel Alloy	Loss of Material/Pitting and Crevice Corrosion	Water Chemistry	Not Applicable
	Nickel Alloy	Loss of Material/Wear	ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD	Not Applicable
Lower Internal Assembly (core support dome)	Stainless Steel	Changes in Dimensions/Void Swelling	PWR Vessel Internals	Note 1
	Stainless Steel	Cracking/Stress Corrosion Cracking, Irradiation-Assisted Stress Corrosion Cracking	PWR Vessel Internals	Note 1
	Stainless Steel	Cracking/Stress Corrosion Cracking, Irradiation-Assisted Stress Corrosion Cracking	Water Chemistry	Note 1
	Stainless Steel	Loss of Fracture Toughness/Neutron Irradiation Embrittlement, Void Swelling	PWR Vessel Internals	Note 1
	Stainless Steel	Loss of Material/Pitting and Crevice Corrosion	Water Chemistry	Note 1

Table B-1 (cont.) Summary of SGS LRA Table 3.1.2-3 – Reactor Vessel Internals Summary of Aging Management Evaluation

Component Type	Material	Aging Effect Requiring Management	Aging Management Programs	Comments
Lower Internal Assembly (core support, incl'g core support lugs, columns and sleeves)	Nickel Alloy	Changes in Dimensions/Void Swelling	PWR Vessel Internals	Note 4
	Nickel Alloy	Cracking/Stress Corrosion Cracking, Irradiation-Assisted Stress Corrosion Cracking	PWR Vessel Internals	Note 4
	Nickel Alloy	Cracking/Stress Corrosion Cracking, Irradiation-Assisted Stress Corrosion Cracking	Water Chemistry	Note 4
	Nickel Alloy	Cumulative Fatigue Damage/Fatigue	TLAA	Note 4
	Nickel Alloy	Loss of Material/Pitting and Crevice Corrosion	Water Chemistry	Note 4
Lower Internal Assembly (core support, incl'g core support lugs, columns and sleeves)	Stainless Steel	Changes in Dimensions/Void Swelling	PWR Vessel Internals	Not Applicable
	Stainless Steel	Cracking/Stress Corrosion Cracking, Irradiation-Assisted Stress Corrosion Cracking	PWR Vessel Internals	Not Applicable
	Stainless Steel	Cracking/Stress Corrosion Cracking, Irradiation-Assisted Stress Corrosion Cracking	Water Chemistry	Not Applicable
	Stainless Steel	Loss of Fracture Toughness/Neutron Irradiation Embrittlement, Void Swelling	PWR Vessel Internals	Not Applicable
	Stainless Steel	Loss of Material/Pitting and Crevice Corrosion	Water Chemistry	Not Applicable

Table B-1 Summary of SGS LRA Table 3.1.2-3 – Reactor Vessel Internals Summary of Aging Management Evaluation

(cont.)

Component Type	Material	Aging Effect Requiring Management	Aging Management Programs	Comments
Lower Internal Assembly (flow distributor (diffuser) plate)	Stainless Steel	Changes in Dimensions/Void Swelling	PWR Vessel Internals	Not Applicable
	Stainless Steel	Cracking/Stress Corrosion Cracking, Irradiation-Assisted Stress Corrosion Cracking	PWR Vessel Internals	Not Applicable
	Stainless Steel	Cracking/Stress Corrosion Cracking, Irradiation-Assisted Stress Corrosion Cracking	Water Chemistry	Not Applicable
	Stainless Steel	Loss of Fracture Toughness/Neutron Irradiation Embrittlement, Void Swelling	PWR Vessel Internals	Not Applicable
	Stainless Steel	Loss of Material/Pitting and Crevice Corrosion	Water Chemistry	Not Applicable
Lower Internal Assembly (fuel assembly locating pin bolts)	Stainless Steel Bolting	Changes in Dimensions/Void Swelling	PWR Vessel Internals	Not Applicable
	Stainless Steel Bolting	Cracking/Stress Corrosion Cracking, Irradiation-Assisted Stress Corrosion Cracking	PWR Vessel Internals	Not Applicable
	Stainless Steel Bolting	Cracking/Stress Corrosion Cracking, Irradiation-Assisted Stress Corrosion Cracking	Water Chemistry	Not Applicable
	Stainless Steel Bolting	Loss of Fracture Toughness/Neutron Irradiation Embrittlement, Void Swelling	PWR Vessel Internals	Not Applicable
	Stainless Steel Bolting	Loss of Material/Pitting and Crevice Corrosion	Water Chemistry	Not Applicable
	Stainless Steel Bolting	Loss of Preload/Stress Relaxation	PWR Vessel Internals	Not Applicable

Table B-1 Summary of SGS LRA Table 3.1.2-3 – Reactor Vessel Internals Summary of Aging Management Evaluation (cont.)

Component Type	Material	Aging Effect Requiring Management	Aging Management Programs	Comments
Lower Internal Assembly (fuel assembly locating pins and lockcaps)	Stainless Steel	Changes in Dimensions/Void Swelling	PWR Vessel Internals	Not Applicable
	Stainless Steel	Cracking/Stress Corrosion Cracking, Irradiation-Assisted Stress Corrosion Cracking	PWR Vessel Internals	Not Applicable
	Stainless Steel	Cracking/Stress Corrosion Cracking, Irradiation-Assisted Stress Corrosion Cracking	Water Chemistry	Not Applicable
	Stainless Steel	Loss of Fracture Toughness/Neutron Irradiation Embrittlement, Void Swelling	PWR Vessel Internals	Not Applicable
	Stainless Steel	Loss of Material/Pitting and Crevice Corrosion	Water Chemistry	Not Applicable
Lower Internal Assembly (inserts for clevis blocks, incl'g lock bars and dowels)	Nickel Alloy	Changes in Dimensions/Void Swelling	PWR Vessel Internals	Not Applicable
	Nickel Alloy	Cracking/Stress Corrosion Cracking, Irradiation-Assisted Stress Corrosion Cracking	PWR Vessel Internals	Not Applicable
	Nickel Alloy	Cracking/Stress Corrosion Cracking, Irradiation-Assisted Stress Corrosion Cracking	Water Chemistry	Not Applicable
	Nickel Alloy	Loss of Material/Pitting and Crevice Corrosion	Water Chemistry	Not Applicable
	Nickel Alloy	Loss of Material/Wear	ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD	Not Applicable

Table B-1 Summary of SGS LRA Table 3.1.2-3 – Reactor Vessel Internals Summary of Aging Management Evaluation

(cont.)

Component Type	Material	Aging Effect Requiring Management	Aging Management Programs	Comments
Lower Internal Assembly (irradiation sample access plugs)	Stainless Steel	Changes in Dimensions/Void Swelling	PWR Vessel Internals	Not Applicable
	Stainless Steel	Cracking/Stress Corrosion Cracking, Irradiation-Assisted Stress Corrosion Cracking	PWR Vessel Internals	Not Applicable
	Stainless Steel	Cracking/Stress Corrosion Cracking, Irradiation-Assisted Stress Corrosion Cracking	Water Chemistry	Not Applicable
	Stainless Steel	Loss of Fracture Toughness/Neutron Irradiation Embrittlement, Void Swelling	PWR Vessel Internals	Not Applicable
	Stainless Steel	Loss of Material/Pitting and Crevice Corrosion	Water Chemistry	Not Applicable
	Stainless Steel	Loss of Preload/Stress Relaxation	PWR Vessel Internals	Not Applicable
Lower Internal Assembly (irradiation sample guide bolts)	Stainless Steel Bolting	Changes in Dimensions/Void Swelling	PWR Vessel Internals	Note 2
	Stainless Steel Bolting	Cracking/Stress Corrosion Cracking, Irradiation-Assisted Stress Corrosion Cracking	PWR Vessel Internals	Note 2
	Stainless Steel Bolting	Cracking/Stress Corrosion Cracking, Irradiation-Assisted Stress Corrosion Cracking	Water Chemistry	Note 2
	Stainless Steel Bolting	Loss of Fracture Toughness/Neutron Irradiation Embrittlement, Void Swelling	PWR Vessel Internals	Note 2
	Stainless Steel Bolting	Loss of Material/Pitting and Crevice Corrosion	Water Chemistry	Note 2
	Stainless Steel Bolting	Loss of Preload/Stress Relaxation	PWR Vessel Internals	Note 2

Table B-1 Summary of SGS LRA Table 3.1.2-3 – Reactor Vessel Internals Summary of Aging Management Evaluation

(cont.)

Component Type	Material	Aging Effect Requiring Management	Aging Management Programs	Comments
Lower Internal Assembly (irradiation sample guide lock caps)	Stainless Steel	Changes in Dimensions/Void Swelling	PWR Vessel Internals	Note 2
	Stainless Steel	Cracking/Stress Corrosion Cracking, Irradiation-Assisted Stress Corrosion Cracking	PWR Vessel Internals	Note 2
	Stainless Steel	Cracking/Stress Corrosion Cracking, Irradiation-Assisted Stress Corrosion Cracking	Water Chemistry	Note 2
	Stainless Steel	Loss of Material/Pitting and Crevice Corrosion	Water Chemistry	Note 2
	Stainless Steel	Loss of Material/Wear	ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD	Note 2
Lower Internal Assembly (irradiation sample guide)	Not Applicable	Not Applicable	Not Applicable	Not Applicable
Lower Internal Assembly (lower core plate)	Stainless Steel	Changes in Dimensions/Void Swelling	PWR Vessel Internals	Not Applicable
	Stainless Steel	Cracking/Stress Corrosion Cracking, Irradiation-Assisted Stress Corrosion Cracking	PWR Vessel Internals	Not Applicable
	Stainless Steel	Cracking/Stress Corrosion Cracking, Irradiation-Assisted Stress Corrosion Cracking	Water Chemistry	Not Applicable
	Stainless Steel	Cumulative Fatigue Damage/Fatigue	TLAA	Not Applicable
	Stainless Steel	Loss of Fracture Toughness/Neutron Irradiation Embrittlement, Void Swelling	PWR Vessel Internals	Not Applicable
	Stainless Steel	Loss of Material/Pitting and Crevice Corrosion	Water Chemistry	Not Applicable

Table B-1 Summary of SGS LRA Table 3.1.2-3 – Reactor Vessel Internals Summary of Aging Management Evaluation

(cont.)

Component Type	Material	Aging Effect Requiring Management	Aging Management Programs	Comments
Lower Internal Assembly (lower core support energy absorbers)	Stainless Steel	Changes in Dimensions/Void Swelling	PWR Vessel Internals	Not Applicable
	Stainless Steel	Cracking/Stress Corrosion Cracking, Irradiation-Assisted Stress Corrosion Cracking	PWR Vessel Internals	Not Applicable
	Stainless Steel	Cracking/Stress Corrosion Cracking, Irradiation-Assisted Stress Corrosion Cracking	Water Chemistry	Not Applicable
	Stainless Steel	Loss of Fracture Toughness/Neutron Irradiation Embrittlement, Void Swelling	PWR Vessel Internals	Not Applicable
	Stainless Steel	Loss of Material/Pitting and Crevice Corrosion	Water Chemistry	Not Applicable
Lower Internal Assembly (lower core support guide post and housing)	Stainless Steel	Changes in Dimensions/Void Swelling	PWR Vessel Internals	Not Applicable
	Stainless Steel	Cracking/Stress Corrosion Cracking, Irradiation-Assisted Stress Corrosion Cracking	PWR Vessel Internals	Not Applicable
	Stainless Steel	Cracking/Stress Corrosion Cracking, Irradiation-Assisted Stress Corrosion Cracking	Water Chemistry	Not Applicable
	Stainless Steel	Loss of Fracture Toughness/Neutron Irradiation Embrittlement, Void Swelling	PWR Vessel Internals	Not Applicable
	Stainless Steel	Loss of Material/Pitting and Crevice Corrosion	Water Chemistry	Not Applicable

Table B-1 (cont.) Summary of SGS LRA Table 3.1.2-3 – Reactor Vessel Internals Summary of Aging Management Evaluation

Component Type	Material	Aging Effect Requiring Management	Aging Management Programs	Comments
Lower Internal Assembly (lower core support ring)	Stainless Steel	Changes in Dimensions/Void Swelling	PWR Vessel Internals	Not Applicable
	Stainless Steel	Cracking/Stress Corrosion Cracking, Irradiation-Assisted Stress Corrosion Cracking	PWR Vessel Internals	Not Applicable
	Stainless Steel	Cracking/Stress Corrosion Cracking, Irradiation-Assisted Stress Corrosion Cracking	Water Chemistry	Not Applicable
	Stainless Steel	Loss of Fracture Toughness/Neutron Irradiation Embrittlement, Void Swelling	PWR Vessel Internals	Not Applicable
	Stainless Steel	Loss of Material/Pitting and Crevice Corrosion	Water Chemistry	Not Applicable
Lower Internal Assembly (lower radial support keys)	Stainless Steel	Changes in Dimensions/Void Swelling	PWR Vessel Internals	Not Applicable
	Stainless Steel	Cracking/Stress Corrosion Cracking, Irradiation-Assisted Stress Corrosion Cracking	PWR Vessel Internals	Not Applicable
	Stainless Steel	Cracking/Stress Corrosion Cracking, Irradiation-Assisted Stress Corrosion Cracking	Water Chemistry	Not Applicable
	Stainless Steel	Loss of Material/Pitting and Crevice Corrosion	Water Chemistry	Not Applicable
	Stainless Steel	Loss of Material/Wear	ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD	Not Applicable

Table B-1 Summary of SGS LRA Table 3.1.2-3 – Reactor Vessel Internals Summary of Aging Management Evaluation

(cont.)

Component Type	Material	Aging Effect Requiring Management	Aging Management Programs	Comments
Lower Internal Assembly (lower support base bolt lock keys)	Stainless Steel	Changes in Dimensions/Void Swelling	PWR Vessel Internals	Not Applicable
	Stainless Steel	Cracking/Stress Corrosion Cracking, Irradiation-Assisted Stress Corrosion Cracking	PWR Vessel Internals	Not Applicable
	Stainless Steel	Cracking/Stress Corrosion Cracking, Irradiation-Assisted Stress Corrosion Cracking	Water Chemistry	Not Applicable
	Stainless Steel	Loss of Material/Pitting and Crevice Corrosion	Water Chemistry	Not Applicable
	Stainless Steel	Loss of Material/Wear	ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD	Not Applicable
Lower Internal Assembly (lower support base bolts)	Stainless Steel Bolting	Changes in Dimensions/Void Swelling	PWR Vessel Internals	Not Applicable
	Stainless Steel Bolting	Cracking/Stress Corrosion Cracking, Irradiation-Assisted Stress Corrosion Cracking	PWR Vessel Internals	Not Applicable
	Stainless Steel Bolting	Cracking/Stress Corrosion Cracking, Irradiation-Assisted Stress Corrosion Cracking	Water Chemistry	Not Applicable
	Stainless Steel Bolting	Loss of Fracture Toughness/Neutron Irradiation Embrittlement, Void Swelling	PWR Vessel Internals	Not Applicable
	Stainless Steel Bolting	Loss of Material/Pitting and Crevice Corrosion	Water Chemistry	Not Applicable
	Stainless Steel Bolting	Loss of Preload/Stress Relaxation	PWR Vessel Internals	Not Applicable

Table B-1 Summary of SGS LRA Table 3.1.2-3 – Reactor Vessel Internals Summary of Aging Management Evaluation (cont.)

Component Type	Material	Aging Effect Requiring Management	Aging Management Programs	Comments
Lower Internal Assembly (lower support column bolts)	Stainless Steel Bolting	Changes in Dimensions/Void Swelling	PWR Vessel Internals	Not Applicable
	Stainless Steel Bolting	Cracking/Stress Corrosion Cracking, Irradiation-Assisted Stress Corrosion Cracking	PWR Vessel Internals	Not Applicable
	Stainless Steel Bolting	Cracking/Stress Corrosion Cracking, Irradiation-Assisted Stress Corrosion Cracking	Water Chemistry	Not Applicable
	Stainless Steel Bolting	Loss of Fracture Toughness/Neutron Irradiation Embrittlement, Void Swelling	PWR Vessel Internals	Not Applicable
	Stainless Steel Bolting	Loss of Preload/Stress Relaxation	PWR Vessel Internals	Not Applicable
Lower Internal Assembly (lower support column nuts)	Stainless Steel	Changes in Dimensions/Void Swelling	PWR Vessel Internals	Not Applicable
	Stainless Steel	Cracking/Stress Corrosion Cracking, Irradiation-Assisted Stress Corrosion Cracking	PWR Vessel Internals	Not Applicable
	Stainless Steel	Cracking/Stress Corrosion Cracking, Irradiation-Assisted Stress Corrosion Cracking	Water Chemistry	Not Applicable
	Stainless Steel	Loss of Fracture Toughness/Neutron Irradiation Embrittlement, Void Swelling	PWR Vessel Internals	Not Applicable
	Stainless Steel	Loss of Material/Pitting and Crevice Corrosion	Water Chemistry	Not Applicable

Table B-1 (cont.) Summary of SGS LRA Table 3.1.2-3 – Reactor Vessel Internals Summary of Aging Management Evaluation

Component Type	Material	Aging Effect Requiring Management	Aging Management Programs	Comments
Lower Internal Assembly (lower support lock keys)	Stainless Steel	Changes in Dimensions/Void Swelling	PWR Vessel Internals	Note 7
	Stainless Steel	Cracking/Stress Corrosion Cracking, Irradiation-Assisted Stress Corrosion Cracking	PWR Vessel Internals	Note 7
	Stainless Steel	Cracking/Stress Corrosion Cracking, Irradiation-Assisted Stress Corrosion Cracking	Water Chemistry	Note 7
	Stainless Steel	Loss of Material/Pitting and Crevice Corrosion	Water Chemistry	Note 7
	Stainless Steel	Loss of Material/Wear	ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD	Note 7
RCCA Guide Tube Assemblies (bolts)	Stainless Steel Bolting	Changes in Dimensions/Void Swelling	PWR Vessel Internals	Not Applicable
	Stainless Steel Bolting	Cracking/Stress Corrosion Cracking, Irradiation-Assisted Stress Corrosion Cracking	PWR Vessel Internals	Not Applicable
	Stainless Steel Bolting	Cracking/Stress Corrosion Cracking, Irradiation-Assisted Stress Corrosion Cracking	Water Chemistry	Not Applicable
	Stainless Steel Bolting	Loss of Material/Pitting and Crevice Corrosion	Water Chemistry	Not Applicable
	Stainless Steel Bolting	Loss of Preload/Stress Relaxation	PWR Vessel Internals	Not Applicable

Table B-1 Summary of SGS LRA Table 3.1.2-3 – Reactor Vessel Internals Summary of Aging Management Evaluation (cont.)

Component Type	Material	Aging Effect Requiring Management	Aging Management Programs	Comments
RCCA Guide Tube Assemblies (enclosures)	Stainless Steel	Changes in Dimensions/Void Swelling	PWR Vessel Internals	Not Applicable
	Stainless Steel	Cracking/Stress Corrosion Cracking, Irradiation-Assisted Stress Corrosion Cracking	PWR Vessel Internals	Not Applicable
	Stainless Steel	Cracking/Stress Corrosion Cracking, Irradiation-Assisted Stress Corrosion Cracking	Water Chemistry	Not Applicable
	Stainless Steel	Loss of Material/Pitting and Crevice Corrosion	Water Chemistry	Not Applicable
RCCA guide tube assemblies (flexures; inserts)	Nickel Alloy	Changes in Dimensions/Void Swelling	PWR Vessel Internals	Note 2
	Nickel Alloy	Cracking/Stress Corrosion Cracking, Irradiation-Assisted Stress Corrosion Cracking	PWR Vessel Internals	Note 2
	Nickel Alloy	Cracking/Stress Corrosion Cracking, Irradiation-Assisted Stress Corrosion Cracking	Water Chemistry	Note 2
	Nickel Alloy	Loss of Material/Pitting and Crevice Corrosion	Water Chemistry	Note 2
	Nickel Alloy	Loss of Material/Wear	ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD	Note 2

Table B-1 (cont.) Summary of SGS LRA Table 3.1.2-3 – Reactor Vessel Internals Summary of Aging Management Evaluation

Component Type	Material	Aging Effect Requiring Management	Aging Management Programs	Comments
RCCA guide tube assemblies (guide pins in tubes)	Stainless Steel	Changes in Dimensions/Void Swelling	PWR Vessel Internals	Not Applicable
	Stainless Steel	Cracking/Stress Corrosion Cracking, Irradiation-Assisted Stress Corrosion Cracking	PWR Vessel Internals	Not Applicable
	Stainless Steel	Cracking/Stress Corrosion Cracking, Irradiation-Assisted Stress Corrosion Cracking	Water Chemistry	Not Applicable
	Stainless Steel	Loss of Material/Pitting and Crevice Corrosion	Water Chemistry	Not Applicable
RCCA guide tube assemblies (lock bars)	Stainless Steel	Changes in Dimensions/Void Swelling	PWR Vessel Internals	Note 2
	Stainless Steel	Cracking/Stress Corrosion Cracking, Irradiation-Assisted Stress Corrosion Cracking	PWR Vessel Internals	Note 2
	Stainless Steel	Cracking/Stress Corrosion Cracking, Irradiation-Assisted Stress Corrosion Cracking	Water Chemistry	Note 2
	Stainless Steel	Loss of Material/Pitting and Crevice Corrosion	Water Chemistry	Note 2
RCCA guide tube assemblies (lower flanges)	Cast Austenitic Stainless Steel (CASS)	Changes in Dimensions/Void Swelling	PWR Vessel Internals	Note 5
	Cast Austenitic Stainless Steel (CASS)	Cracking/Stress Corrosion Cracking, Irradiation-Assisted Stress Corrosion Cracking	PWR Vessel Internals	Note 5
	Cast Austenitic Stainless Steel (CASS)	Cracking/Stress Corrosion Cracking, Irradiation-Assisted Stress Corrosion Cracking	Water Chemistry	Note 5
	Cast Austenitic Stainless Steel (CASS)	Loss of Material/Pitting and Crevice Corrosion	Water Chemistry	Note 5

Table B-1 Summary of SGS LRA Table 3.1.2-3 – Reactor Vessel Internals Summary of Aging Management Evaluation

(cont.)

Component Type	Material	Aging Effect Requiring Management	Aging Management Programs	Comments
RCCA guide tube assemblies (pins, anti-rotation studs, and nuts)	Stainless Steel	Changes in Dimensions/Void Swelling	PWR Vessel Internals	Note 2
	Stainless Steel	Cracking/Stress Corrosion Cracking, Irradiation-Assisted Stress Corrosion Cracking	PWR Vessel Internals	Note 2
	Stainless Steel	Cracking/Stress Corrosion Cracking, Irradiation-Assisted Stress Corrosion Cracking	Water Chemistry	Note 2
	Stainless Steel	Loss of Material/Pitting and Crevice Corrosion	Water Chemistry	Note 2
RCCA guide tube assemblies (sheaths)	Stainless Steel	Changes in Dimensions/Void Swelling	PWR Vessel Internals	Not Applicable
	Stainless Steel	Cracking/Stress Corrosion Cracking, Irradiation-Assisted Stress Corrosion Cracking	PWR Vessel Internals	Not Applicable
	Stainless Steel	Cracking/Stress Corrosion Cracking, Irradiation-Assisted Stress Corrosion Cracking	Water Chemistry	Not Applicable
	Stainless Steel	Loss of Material/Pitting and Crevice Corrosion	Water Chemistry	Not Applicable
RCCA guide tube assemblies (support pin cover plate)	Stainless Steel	Changes in Dimensions/Void Swelling	PWR Vessel Internals	Note 2
	Stainless Steel	Cracking/Stress Corrosion Cracking, Irradiation-Assisted Stress Corrosion Cracking	PWR Vessel Internals	Note 2
	Stainless Steel	Cracking/Stress Corrosion Cracking, Irradiation-Assisted Stress Corrosion Cracking	Water Chemistry	Note 2
	Stainless Steel	Loss of Material/Pitting and Crevice Corrosion	Water Chemistry	Note 2

Table B-1 Summary of SGS LRA Table 3.1.2-3 – Reactor Vessel Internals Summary of Aging Management Evaluation (cont.)

Component Type	Material	Aging Effect Requiring Management	Aging Management Programs	Comments
RCCA guide tube assemblies (support pin fasteners and nuts)	Stainless Steel	Changes in Dimensions/Void Swelling	PWR Vessel Internals	Not Applicable
	Stainless Steel	Cracking/Stress Corrosion Cracking, Irradiation-Assisted Stress Corrosion Cracking	PWR Vessel Internals	Not Applicable
	Stainless Steel	Cracking/Stress Corrosion Cracking, Irradiation-Assisted Stress Corrosion Cracking	Water Chemistry	Not Applicable
	Stainless Steel	Loss of Material/Pitting and Crevice Corrosion	Water Chemistry	Not Applicable
RCCA guide tube assemblies (tubes, housing plates, and guide plates)	Stainless Steel	Changes in Dimensions/Void Swelling	PWR Vessel Internals	Note 5
	Stainless Steel	Cracking/Stress Corrosion Cracking, Irradiation-Assisted Stress Corrosion Cracking	PWR Vessel Internals	Note 5
	Stainless Steel	Cracking/Stress Corrosion Cracking, Irradiation-Assisted Stress Corrosion Cracking	Water Chemistry	Note 5
	Stainless Steel	Loss of Material/Pitting and Crevice Corrosion	Water Chemistry	Note 5
RCCA guide tube assemblies (upper guide tube)	Stainless Steel	Changes in Dimensions/Void Swelling	PWR Vessel Internals	Not Applicable
	Stainless Steel	Cracking/Stress Corrosion Cracking, Irradiation-Assisted Stress Corrosion Cracking	PWR Vessel Internals	Not Applicable
	Stainless Steel	Cracking/Stress Corrosion Cracking, Irradiation-Assisted Stress Corrosion Cracking	Water Chemistry	Not Applicable
	Stainless Steel	Loss of Material/Pitting and Crevice Corrosion	Water Chemistry	Not Applicable

Table B-1 Summary of SGS LRA Table 3.1.2-3 – Reactor Vessel Internals Summary of Aging Management Evaluation

(cont.)

Component Type	Material	Aging Effect Requiring Management	Aging Management Programs	Comments
Reactor Vessel Internals (core support locking nut)	Stainless Steel	Changes in Dimensions/Void Swelling	PWR Vessel Internals	Not Applicable
	Stainless Steel	Cracking/Stress Corrosion Cracking, Irradiation-Assisted Stress Corrosion Cracking	PWR Vessel Internals	Not Applicable
	Stainless Steel	Cracking/Stress Corrosion Cracking, Irradiation-Assisted Stress Corrosion Cracking	Water Chemistry	Not Applicable
	Stainless Steel	Loss of Fracture Toughness/Neutron Irradiation Embrittlement, Void Swelling	PWR Vessel Internals	Not Applicable
	Stainless Steel	Loss of Material/Pitting and Crevice Corrosion	Water Chemistry	Not Applicable
Reactor Vessel Internals (flux thimbles - tubes)	Stainless Steel	Changes in Dimensions/Void Swelling	PWR Vessel Internals	Note 6
	Stainless Steel	Cracking/Stress Corrosion Cracking, Irradiation-Assisted Stress Corrosion Cracking	PWR Vessel Internals	Note 6
	Stainless Steel	Cracking/Stress Corrosion Cracking, Irradiation-Assisted Stress Corrosion Cracking	Water Chemistry	Note 6
	Stainless Steel	Loss of Material/Pitting and Crevice Corrosion	Water Chemistry	Note 6
	Stainless Steel	Loss of Material/Wear	Flux Thimble Tube Inspection	Note 6

Table B-1 Summary of SGS LRA Table 3.1.2-3 – Reactor Vessel Internals Summary of Aging Management Evaluation

(cont.)

Component Type	Material	Aging Effect Requiring Management	Aging Management Programs	Comments
Reactor Vessel Internals (incore guide cruciforms)	Cast Austenitic Stainless Steel (CASS)	Changes in Dimensions/Void Swelling	PWR Vessel Internals	Note 3
	Cast Austenitic Stainless Steel (CASS)	Cracking/Stress Corrosion Cracking, Irradiation-Assisted Stress Corrosion Cracking	PWR Vessel Internals	Note 3
	Cast Austenitic Stainless Steel (CASS)	Cracking/Stress Corrosion Cracking, Irradiation-Assisted Stress Corrosion Cracking	Water Chemistry	Note 3
	Cast Austenitic Stainless Steel (CASS)	Loss of Fracture Toughness/Neutron Irradiation Embrittlement, Void Swelling	PWR Vessel Internals	Note 3
	Cast Austenitic Stainless Steel (CASS)	Loss of Fracture Toughness/Thermal Aging and Neutron Irradiation Embrittlement	PWR Vessel Internals	Note 3
	Cast Austenitic Stainless Steel (CASS)	Loss of Material/Pitting and Crevice Corrosion	Water Chemistry	Note 3
Reactor Vessel Internals (incore guide tube column bodies)	Stainless Steel	Changes in Dimensions/Void Swelling	PWR Vessel Internals	Not Applicable
	Stainless Steel	Cracking/Stress Corrosion Cracking, Irradiation-Assisted Stress Corrosion Cracking	PWR Vessel Internals	Not Applicable
	Stainless Steel	Cracking/Stress Corrosion Cracking, Irradiation-Assisted Stress Corrosion Cracking	Water Chemistry	Not Applicable
	Stainless Steel	Loss of Fracture Toughness/Neutron Irradiation Embrittlement, Void Swelling	PWR Vessel Internals	Not Applicable
	Stainless Steel	Loss of Material/Pitting and Crevice Corrosion	Water Chemistry	Not Applicable

Table B-1 Summary of SGS LRA Table 3.1.2-3 – Reactor Vessel Internals Summary of Aging Management Evaluation

(cont.)

Component Type	Material	Aging Effect Requiring Management	Aging Management Programs	Comments
Reactor Vessel Internals (incore guide tube extensions)	Stainless Steel	Changes in Dimensions/Void Swelling	PWR Vessel Internals	Not Applicable
	Stainless Steel	Cracking/Stress Corrosion Cracking, Irradiation-Assisted Stress Corrosion Cracking	PWR Vessel Internals	Not Applicable
	Stainless Steel	Cracking/Stress Corrosion Cracking, Irradiation-Assisted Stress Corrosion Cracking	Water Chemistry	Not Applicable
	Stainless Steel	Loss of Fracture Toughness/Neutron Irradiation Embrittlement, Void Swelling	PWR Vessel Internals	Not Applicable
	Stainless Steel	Loss of Material/Pitting and Crevice Corrosion	Water Chemistry	Not Applicable
Reactor Vessel Internals (incore instrument guide extension bolt lock caps)	Stainless Steel	Changes in Dimensions/Void Swelling	PWR Vessel Internals	Not Applicable
	Stainless Steel	Cracking/Stress Corrosion Cracking, Irradiation-Assisted Stress Corrosion Cracking	PWR Vessel Internals	Not Applicable
	Stainless Steel	Cracking/Stress Corrosion Cracking, Irradiation-Assisted Stress Corrosion Cracking	Water Chemistry	Not Applicable
	Stainless Steel	Loss of Fracture Toughness/Neutron Irradiation Embrittlement, Void Swelling	PWR Vessel Internals	Not Applicable
	Stainless Steel	Loss of Material/Pitting and Crevice Corrosion	Water Chemistry	Not Applicable

Table B-1 Summary of SGS LRA Table 3.1.2-3 – Reactor Vessel Internals Summary of Aging Management Evaluation

(cont.)

Component Type	Material	Aging Effect Requiring Management	Aging Management Programs	Comments
Reactor Vessel Internals (incore instrument guide extension bolts)	Stainless Steel Bolting	Changes in Dimensions/Void Swelling	PWR Vessel Internals	Not Applicable
	Stainless Steel Bolting	Cracking/Stress Corrosion Cracking, Irradiation-Assisted Stress Corrosion Cracking	PWR Vessel Internals	Not Applicable
	Stainless Steel Bolting	Cracking/Stress Corrosion Cracking, Irradiation-Assisted Stress Corrosion Cracking	Water Chemistry	Not Applicable
	Stainless Steel Bolting	Loss of Fracture Toughness/Neutron Irradiation Embrittlement, Void Swelling	PWR Vessel Internals	Not Applicable
	Stainless Steel Bolting	Loss of Material/Pitting and Crevice Corrosion	Water Chemistry	Not Applicable
	Stainless Steel Bolting	Loss of Preload/Stress Relaxation	PWR Vessel Internals	Not Applicable
Reactor Vessel Internals (incore instrument guide extension collars)	Stainless Steel	Changes in Dimensions/Void Swelling	PWR Vessel Internals	Not Applicable
	Stainless Steel	Cracking/Stress Corrosion Cracking, Irradiation-Assisted Stress Corrosion Cracking	PWR Vessel Internals	Not Applicable
	Stainless Steel	Cracking/Stress Corrosion Cracking, Irradiation-Assisted Stress Corrosion Cracking	Water Chemistry	Not Applicable
	Stainless Steel	Loss of Fracture Toughness/Neutron Irradiation Embrittlement, Void Swelling	PWR Vessel Internals	Not Applicable
	Stainless Steel	Loss of Material/Pitting and Crevice Corrosion	Water Chemistry	Not Applicable

Table B-1 (cont.) Summary of SGS LRA Table 3.1.2-3 – Reactor Vessel Internals Summary of Aging Management Evaluation

Component Type	Material	Aging Effect Requiring Management	Aging Management Programs	Comments
Reactor Vessel Internals (incore instrument guide extension nuts)	Stainless Steel	Changes in Dimensions/Void Swelling	PWR Vessel Internals	Not Applicable
	Stainless Steel	Cracking/Stress Corrosion Cracking, Irradiation-Assisted Stress Corrosion Cracking	PWR Vessel Internals	Not Applicable
	Stainless Steel	Cracking/Stress Corrosion Cracking, Irradiation-Assisted Stress Corrosion Cracking	Water Chemistry	Not Applicable
	Stainless Steel	Loss of Fracture Toughness/Neutron Irradiation Embrittlement, Void Swelling	PWR Vessel Internals	Not Applicable
	Stainless Steel	Loss of Material/Pitting and Crevice Corrosion	Water Chemistry	Not Applicable
Reactor Vessel Internals (incore instrument guide tube extension bars)	Stainless Steel	Changes in Dimensions/Void Swelling	PWR Vessel Internals	Not Applicable
	Stainless Steel	Cracking/Stress Corrosion Cracking, Irradiation-Assisted Stress Corrosion Cracking	PWR Vessel Internals	Not Applicable
	Stainless Steel	Cracking/Stress Corrosion Cracking, Irradiation-Assisted Stress Corrosion Cracking	Water Chemistry	Not Applicable
	Stainless Steel	Loss of Fracture Toughness/Neutron Irradiation Embrittlement, Void Swelling	PWR Vessel Internals	Not Applicable
	Stainless Steel	Loss of Material/Pitting and Crevice Corrosion	Water Chemistry	Not Applicable

Table B-1 (cont.) Summary of SGS LRA Table 3.1.2-3 – Reactor Vessel Internals Summary of Aging Management Evaluation

Component Type	Material	Aging Effect Requiring Management	Aging Management Programs	Comments
Reactor Vessel Internals (manway cover assembly)	Stainless Steel	Changes in Dimensions/Void Swelling	PWR Vessel Internals	Not Applicable
	Stainless Steel	Cracking/Stress Corrosion Cracking, Irradiation-Assisted Stress Corrosion Cracking	PWR Vessel Internals	Not Applicable
	Stainless Steel	Cracking/Stress Corrosion Cracking, Irradiation-Assisted Stress Corrosion Cracking	Water Chemistry	Not Applicable
	Stainless Steel	Loss of Fracture Toughness/Neutron Irradiation Embrittlement, Void Swelling	PWR Vessel Internals	Not Applicable
	Stainless Steel	Loss of Material/Pitting and Crevice Corrosion	Water Chemistry	Not Applicable
Reactor Vessel Internals (secondary core support)	Stainless Steel	Changes in Dimensions/Void Swelling	PWR Vessel Internals	Not Applicable
	Stainless Steel	Cracking/Stress Corrosion Cracking, Irradiation-Assisted Stress Corrosion Cracking	PWR Vessel Internals	Not Applicable
	Stainless Steel	Cracking/Stress Corrosion Cracking, Irradiation-Assisted Stress Corrosion Cracking	Water Chemistry	Not Applicable
	Stainless Steel	Loss of Fracture Toughness/Neutron Irradiation Embrittlement, Void Swelling	PWR Vessel Internals	Not Applicable
	Stainless Steel	Loss of Material/Pitting and Crevice Corrosion	Water Chemistry	Not Applicable

Table B-1 Summary of SGS LRA Table 3.1.2-3 – Reactor Vessel Internals Summary of Aging Management Evaluation

(cont.)

Component Type	Material	Aging Effect Requiring Management	Aging Management Programs	Comments
Thermal Shield	Stainless Steel	Changes in Dimensions/Void Swelling	PWR Vessel Internals	Not Applicable
	Stainless Steel	Cracking/Stress Corrosion Cracking, Irradiation-Assisted Stress Corrosion Cracking	PWR Vessel Internals	Not Applicable
	Stainless Steel	Cracking/Stress Corrosion Cracking, Irradiation-Assisted Stress Corrosion Cracking	Water Chemistry	Not Applicable
	Stainless Steel	Loss of Fracture Toughness/Neutron Irradiation Embrittlement, Void Swelling	PWR Vessel Internals	Not Applicable
	Stainless Steel	Loss of Material/Pitting and Crevice Corrosion	Water Chemistry	Not Applicable
Thermal Shield (adjustment plugs)	Stainless Steel	Changes in Dimensions/Void Swelling	PWR Vessel Internals	Not Applicable
	Stainless Steel	Cracking/Stress Corrosion Cracking, Irradiation-Assisted Stress Corrosion Cracking	PWR Vessel Internals	Not Applicable
	Stainless Steel	Cracking/Stress Corrosion Cracking, Irradiation-Assisted Stress Corrosion Cracking	Water Chemistry	Not Applicable
	Stainless Steel	Loss of Fracture Toughness/Neutron Irradiation Embrittlement, Void Swelling	PWR Vessel Internals	Not Applicable
	Stainless Steel	Loss of Material/Pitting and Crevice Corrosion	Water Chemistry	Not Applicable

Table B-1 Summary of SGS LRA Table 3.1.2-3 – Reactor Vessel Internals Summary of Aging Management Evaluation

(cont.)

Component Type	Material	Aging Effect Requiring Management	Aging Management Programs	Comments
Thermal Shield (bolts and dowels)	Stainless Steel Bolting	Changes in Dimensions/Void Swelling	PWR Vessel Internals	Not Applicable
	Stainless Steel Bolting	Cracking/Stress Corrosion Cracking, Irradiation-Assisted Stress Corrosion Cracking	PWR Vessel Internals	Not Applicable
	Stainless Steel Bolting	Cracking/Stress Corrosion Cracking, Irradiation-Assisted Stress Corrosion Cracking	Water Chemistry	Not Applicable
	Stainless Steel Bolting	Loss of Fracture Toughness/Neutron Irradiation Embrittlement, Void Swelling	PWR Vessel Internals	Not Applicable
	Stainless Steel Bolting	Loss of Material/Pitting and Crevice Corrosion	Water Chemistry	Not Applicable
	Stainless Steel Bolting	Loss of Preload/Stress Relaxation	PWR Vessel Internals	Not Applicable
Upper Internals Assembly (beam and ribs bolts)	Stainless Steel Bolting	Changes in Dimensions/Void Swelling	PWR Vessel Internals	Not Applicable
	Stainless Steel Bolting	Cracking/Stress Corrosion Cracking, Irradiation-Assisted Stress Corrosion Cracking	PWR Vessel Internals	Not Applicable
	Stainless Steel Bolting	Cracking/Stress Corrosion Cracking, Irradiation-Assisted Stress Corrosion Cracking	Water Chemistry	Not Applicable
	Stainless Steel Bolting	Loss of Material/Pitting and Crevice Corrosion	Water Chemistry	Not Applicable
	Stainless Steel Bolting	Loss of Preload/Stress Relaxation	PWR Vessel Internals	Not Applicable

Table B-1 Summary of SGS LRA Table 3.1.2-3 – Reactor Vessel Internals Summary of Aging Management Evaluation (cont.)

Component Type	Material	Aging Effect Requiring Management	Aging Management Programs	Comments
Upper Internals Assembly (beam and ribs lock keys)	Stainless Steel	Changes in Dimensions/Void Swelling	PWR Vessel Internals	Not Applicable
	Stainless Steel	Cracking/Stress Corrosion Cracking, Irradiation-Assisted Stress Corrosion Cracking	PWR Vessel Internals	Not Applicable
	Stainless Steel	Cracking/Stress Corrosion Cracking, Irradiation-Assisted Stress Corrosion Cracking	Water Chemistry	Not Applicable
	Stainless Steel	Loss of Material/Pitting and Crevice Corrosion	Water Chemistry	Not Applicable
	Stainless Steel	Loss of Material/Wear	ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD	Not Applicable
Upper Internals Assembly (capped top thermocouple columns)	Stainless Steel	Changes in Dimensions/Void Swelling	PWR Vessel Internals	Note 2
	Stainless Steel	Cracking/Stress Corrosion Cracking, Irradiation-Assisted Stress Corrosion Cracking	PWR Vessel Internals	Note 2
	Stainless Steel	Cracking/Stress Corrosion Cracking, Irradiation-Assisted Stress Corrosion Cracking	Water Chemistry	Note 2
	Stainless Steel	Loss of Material/Pitting and Crevice Corrosion	Water Chemistry	Note 2

Table B-1 Summary of SGS LRA Table 3.1.2-3 – Reactor Vessel Internals Summary of Aging Management Evaluation

(cont.)

Component Type	Material	Aging Effect Requiring Management	Aging Management Programs	Comments
Upper Internals Assembly (deep beam rib and stiffener, and ribs)	Stainless Steel	Changes in Dimensions/Void Swelling	PWR Vessel Internals	Not Applicable
	Stainless Steel	Cracking/Stress Corrosion Cracking, Irradiation-Assisted Stress Corrosion Cracking	PWR Vessel Internals	Not Applicable
	Stainless Steel	Cracking/Stress Corrosion Cracking, Irradiation-Assisted Stress Corrosion Cracking	Water Chemistry	Not Applicable
	Stainless Steel	Loss of Material/Pitting and Crevice Corrosion	Water Chemistry	Not Applicable
Upper Internals Assembly (fuel assembly locating pins)	Stainless Steel	Changes in Dimensions/Void Swelling	PWR Vessel Internals	Not Applicable
	Stainless Steel	Cracking/Stress Corrosion Cracking, Irradiation-Assisted Stress Corrosion Cracking	PWR Vessel Internals	Not Applicable
	Stainless Steel	Cracking/Stress Corrosion Cracking, Irradiation-Assisted Stress Corrosion Cracking	Water Chemistry	Not Applicable
	Stainless Steel	Loss of Material/Pitting and Crevice Corrosion	Water Chemistry	Not Applicable
Upper Internals Assembly (head to vessel alignment pin bolt locking caps)	Stainless Steel	Changes in Dimensions/Void Swelling	PWR Vessel Internals	Note 2
	Stainless Steel	Cracking/Stress Corrosion Cracking, Irradiation-Assisted Stress Corrosion Cracking	PWR Vessel Internals	Note 2
	Stainless Steel	Cracking/Stress Corrosion Cracking, Irradiation-Assisted Stress Corrosion Cracking	Water Chemistry	Note 2
	Stainless Steel	Loss of Material/Pitting and Crevice Corrosion	Water Chemistry	Note 2

Table B-1 Summary of SGS LRA Table 3.1.2-3 – Reactor Vessel Internals Summary of Aging Management Evaluation

(cont.)

Component Type	Material	Aging Effect Requiring Management	Aging Management Programs	Comments
Upper Internals Assembly (head to vessel alignment pin bolts)	Stainless Steel Bolting	Changes in Dimensions/Void Swelling	PWR Vessel Internals	Not Applicable
	Stainless Steel Bolting	Cracking/Stress Corrosion Cracking, Irradiation-Assisted Stress Corrosion Cracking	PWR Vessel Internals	Not Applicable
	Stainless Steel Bolting	Cracking/Stress Corrosion Cracking, Irradiation-Assisted Stress Corrosion Cracking	Water Chemistry	Not Applicable
	Stainless Steel Bolting	Loss of Material/Pitting and Crevice Corrosion	Water Chemistry	Not Applicable
	Stainless Steel Bolting	Loss of Preload/Stress Relaxation	PWR Vessel Internals	Not Applicable
Upper Internals Assembly (head to vessel alignment pins)	Stainless Steel	Changes in Dimensions/Void Swelling	PWR Vessel Internals	Not Applicable
	Stainless Steel	Cracking/Stress Corrosion Cracking, Irradiation-Assisted Stress Corrosion Cracking	PWR Vessel Internals	Not Applicable
	Stainless Steel	Cracking/Stress Corrosion Cracking, Irradiation-Assisted Stress Corrosion Cracking	Water Chemistry	Not Applicable
	Stainless Steel	Loss of Material/Pitting and Crevice Corrosion	Water Chemistry	Not Applicable
	Stainless Steel	Loss of Material/Wear	ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD	Not Applicable

Table B-1 Summary of SGS LRA Table 3.1.2-3 – Reactor Vessel Internals Summary of Aging Management Evaluation

(cont.)

Component Type	Material	Aging Effect Requiring Management	Aging Management Programs	Comments
Upper Internals Assembly (hold down spring)	Stainless Steel	Changes in Dimensions/Void Swelling	PWR Vessel Internals	Not Applicable
	Stainless Steel	Cracking/Stress Corrosion Cracking, Irradiation-Assisted Stress Corrosion Cracking	PWR Vessel Internals	Not Applicable
	Stainless Steel	Cracking/Stress Corrosion Cracking, Irradiation-Assisted Stress Corrosion Cracking	Water Chemistry	Not Applicable
	Stainless Steel	Loss of Material/Pitting and Crevice Corrosion	Water Chemistry	Not Applicable
	Stainless Steel	Loss of Preload/Stress Relaxation	PWR Vessel Internals	Not Applicable
Upper Internals Assembly (nuts)	Stainless Steel	Changes in Dimensions/Void Swelling	PWR Vessel Internals	Not Applicable
	Stainless Steel	Cracking/Stress Corrosion Cracking, Irradiation-Assisted Stress Corrosion Cracking	PWR Vessel Internals	Not Applicable
	Stainless Steel	Cracking/Stress Corrosion Cracking, Irradiation-Assisted Stress Corrosion Cracking	Water Chemistry	Not Applicable
	Stainless Steel	Loss of Material/Pitting and Crevice Corrosion	Water Chemistry	Not Applicable

Table B-1 (cont.) Summary of SGS LRA Table 3.1.2-3 – Reactor Vessel Internals Summary of Aging Management Evaluation

Component Type	Material	Aging Effect Requiring Management	Aging Management Programs	Comments
Upper Internals Assembly (orifice plates)	Stainless Steel	Changes in Dimensions/Void Swelling	PWR Vessel Internals	Note 1
	Stainless Steel	Cracking/Stress Corrosion Cracking, Irradiation-Assisted Stress Corrosion Cracking	PWR Vessel Internals	Note 1
	Stainless Steel	Cracking/Stress Corrosion Cracking, Irradiation-Assisted Stress Corrosion Cracking	Water Chemistry	Note 1
	Stainless Steel	Loss of Material/Pitting and Crevice Corrosion	Water Chemistry	Note 1
Upper Internals Assembly (static flow mixers)	Cast Austenitic Stainless Steel (CASS)	Changes in Dimensions/Void Swelling	PWR Vessel Internals	Not Applicable
	Cast Austenitic Stainless Steel (CASS)	Cracking/Stress Corrosion Cracking, Irradiation-Assisted Stress Corrosion Cracking	PWR Vessel Internals	Not Applicable
	Cast Austenitic Stainless Steel (CASS)	Cracking/Stress Corrosion Cracking, Irradiation-Assisted Stress Corrosion Cracking	Water Chemistry	Not Applicable
	Cast Austenitic Stainless Steel (CASS)	Loss of Material/Pitting and Crevice Corrosion	Water Chemistry	Not Applicable

Table B-1 Summary of SGS LRA Table 3.1.2-3 – Reactor Vessel Internals Summary of Aging Management Evaluation

(cont.)

Component Type	Material	Aging Effect Requiring Management	Aging Management Programs	Comments
Upper Internals Assembly (upper core plate alignment pins)	Stainless Steel	Changes in Dimensions/Void Swelling	PWR Vessel Internals	Not Applicable
	Stainless Steel	Cracking/Stress Corrosion Cracking, Irradiation-Assisted Stress Corrosion Cracking	PWR Vessel Internals	Not Applicable
	Stainless Steel	Cracking/Stress Corrosion Cracking, Irradiation-Assisted Stress Corrosion Cracking	Water Chemistry	Not Applicable
	Stainless Steel	Loss of Material/Pitting and Crevice Corrosion	Water Chemistry	Not Applicable
	Stainless Steel	Loss of Material/Wear	ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD	Not Applicable
Upper Internals Assembly (upper core plate, insert, spacer ring, upper support plate, and upper support ring or skirt)	Stainless Steel	Changes in Dimensions/Void Swelling	PWR Vessel Internals	Not Applicable
	Stainless Steel	Cracking/Stress Corrosion Cracking, Irradiation-Assisted Stress Corrosion Cracking	PWR Vessel Internals	Not Applicable
	Stainless Steel	Cracking/Stress Corrosion Cracking, Irradiation-Assisted Stress Corrosion Cracking	Water Chemistry	Not Applicable
	Stainless Steel	Loss of Material/Pitting and Crevice Corrosion	Water Chemistry	Not Applicable

Table B-1 Summary of SGS LRA Table 3.1.2-3 – Reactor Vessel Internals Summary of Aging Management Evaluation (cont.)

Component Type	Material	Aging Effect Requiring Management	Aging Management Programs	Comments
Upper Internals Assembly (upper support column bases)	Cast Austenitic Stainless Steel (CASS)	Changes in Dimensions/Void Swelling	PWR Vessel Internals	Not Applicable
	Cast Austenitic Stainless Steel (CASS)	Cracking/Stress Corrosion Cracking, Irradiation-Assisted Stress Corrosion Cracking	PWR Vessel Internals	Not Applicable
	Cast Austenitic Stainless Steel (CASS)	Cracking/Stress Corrosion Cracking, Irradiation-Assisted Stress Corrosion Cracking	Water Chemistry	Not Applicable
	Cast Austenitic Stainless Steel (CASS)	Loss of Material/Pitting and Crevice Corrosion	Water Chemistry	Not Applicable
Upper Internals Assembly (upper support column bodies)	Stainless Steel	Changes in Dimensions/Void Swelling	PWR Vessel Internals	Not Applicable
	Stainless Steel	Cracking/Stress Corrosion Cracking, Irradiation-Assisted Stress Corrosion Cracking	PWR Vessel Internals	Not Applicable
	Stainless Steel	Cracking/Stress Corrosion Cracking, Irradiation-Assisted Stress Corrosion Cracking	Water Chemistry	Not Applicable
	Stainless Steel	Loss of Material/Pitting and Crevice Corrosion	Water Chemistry	Not Applicable

Table B-1 Summary of SGS LRA Table 3.1.2-3 – Reactor Vessel Internals Summary of Aging Management Evaluation

(cont.)

Component Type	Material	Aging Effect Requiring Management	Aging Management Programs	Comments
Upper Internals Assembly (upper support column bolts)	Stainless Steel Bolting	Changes in Dimensions/Void Swelling	PWR Vessel Internals	Not Applicable
	Stainless Steel Bolting	Cracking/Stress Corrosion Cracking, Irradiation-Assisted Stress Corrosion Cracking	PWR Vessel Internals	Not Applicable
	Stainless Steel Bolting	Cracking/Stress Corrosion Cracking, Irradiation-Assisted Stress Corrosion Cracking	Water Chemistry	Not Applicable
	Stainless Steel Bolting	Loss of Material/Pitting and Crevice Corrosion	Water Chemistry	Not Applicable
	Stainless Steel Bolting	Loss of Preload/Stress Relaxation	PWR Vessel Internals	Not Applicable
Upper Internals Assembly (upper support column extension tubes and adapters)	Stainless Steel	Changes in Dimensions/Void Swelling	PWR Vessel Internals	Not Applicable
	Stainless Steel	Cracking/Stress Corrosion Cracking, Irradiation-Assisted Stress Corrosion Cracking	PWR Vessel Internals	Not Applicable
	Stainless Steel	Cracking/Stress Corrosion Cracking, Irradiation-Assisted Stress Corrosion Cracking	Water Chemistry	Not Applicable
	Stainless Steel	Loss of Material/Pitting and Crevice Corrosion	Water Chemistry	Not Applicable

Table B-1 Summary of SGS LRA Table 3.1.2-3 – Reactor Vessel Internals Summary of Aging Management Evaluation

(cont.)

Component Type	Material	Aging Effect Requiring Management	Aging Management Programs	Comments
Upper Internals Assembly (upper support column flanges)	Stainless Steel	Changes in Dimensions/Void Swelling	PWR Vessel Internals	Not Applicable
	Stainless Steel	Cracking/Stress Corrosion Cracking, Irradiation-Assisted Stress Corrosion Cracking	PWR Vessel Internals	Not Applicable
	Stainless Steel	Cracking/Stress Corrosion Cracking, Irradiation-Assisted Stress Corrosion Cracking	Water Chemistry	Not Applicable
	Stainless Steel	Loss of Material/Pitting and Crevice Corrosion	Water Chemistry	Not Applicable

Table B-1 Summary of SGS LRA Table 3.1.2-3 – Reactor Vessel Internals Summary of Aging Management Evaluation

(cont.)

Component Type	Material	Aging Effect Requiring Management	Aging Management Programs	Comments
Upper Internals Assembly (upper support column lock keys)	Stainless Steel	Changes in Dimensions/Void Swelling	PWR Vessel Internals	Not Applicable
	Stainless Steel	Cracking/Stress Corrosion Cracking, Irradiation-Assisted Stress Corrosion Cracking	PWR Vessel Internals	Not Applicable
	Stainless Steel	Cracking/Stress Corrosion Cracking, Irradiation-Assisted Stress Corrosion Cracking	Water Chemistry	Not Applicable
	Stainless Steel	Loss of Material/Pitting and Crevice Corrosion	Water Chemistry	Not Applicable
	Stainless Steel	Loss of Material/Wear	ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD	Not Applicable

Note:

1. During the evaluations performed to address A/LAI 1, 2, and 7, it was identified that this component for SGS Unit 2 is fabricated from CASS material. See Section 6.2.7 and Table 6-2 for additional information.
2. These components and associated aging effects requiring management are not applicable to SGS Unit 2. These components were listed within Table 3.1.2-3 of the SGS LRA but do not exist within SGS Unit 2 [41]. These components have been left in this table to maintain consistency with [1].
3. During the evaluations performed to address A/LAI 1 and 2, it was identified that this component for SGS Unit 2 is fabricated from 304 stainless steel. Therefore, this component was not evaluated to address A/LAI 7 in Section 6.2.7.
4. During the evaluations performed to address A/LAI 1 and 2, it was identified that one of the components in this category for SGS Unit 2, specifically the “Lower Support Column – Cap,” is fabricated from CASS material See Section 6.2.7 and Table 6-2 for additional information. No components within this category were identified to be fabricated from nickel alloy.
5. During the evaluations performed to address A/LAI 1, 2, and 7, it was identified that this component for SGS Unit 2 is fabricated from CASS material and 304 stainless steel. See Section 6.2.7 and Table 6-2 for additional information.
6. During the evaluations performed to address A/LAI 1 and 2, it was identified that this component for SGS Unit 2 is fabricated from Alloy 600.
7. This component type is the same as the “Lower Internal Assembly (lower radial support keys)” as identified in the SGS Unit 2 AMR (Lower Internals Assembly – Radial Support Keys per MRP-191 [14]).

APPENDIX C
MRP-227-A AUGMENTED INSPECTIONS

Table C-1 MRP-227-A Primary Inspection and Monitoring Recommendations for Westinghouse-Designed Internals

Item	Applicability	Effect (Mechanism)	Expansion Link (Note 1)	Examination Method/Frequency (Note 1)	Examination Coverage
Control Rod Guide Tube Assembly Guide plates (cards)	All plants	Loss of Material (wear)	None	Visual (VT-3) Per the schedule requirements of WCAP-17451-P Section 5 including subsequent examinations. (Note 7)	Minimum examination of 20% of the number of CRGT assemblies, and as per the requirements of WCAP-17451-P Revision 1 Section 5. (Note 7) See Figure A-2
Control Rod Guide Tube Assembly Lower flange welds	All plants	Cracking (SCC, Fatigue) Aging Management (IE and TE)	Bottom-mounted instrumentation (BMI) column bodies, lower support column bodies (cast), Upper core plate, Lower support forging/casting	Enhanced visual (EVT-1) examination to determine the presence of crack-like surface flaws in flange welds no later than 2 refueling outages from the beginning of the license renewal period and subsequent examination on a ten-year interval.	100% of outer (accessible) CRGT lower flange weld surfaces and adjacent base metal on the individual periphery CRGT assemblies. (Note 2) See Figure A-3.
Core Barrel Assembly Upper core barrel flange weld	All plants	Cracking (SCC)	Lower support column bodies (non-cast) Core barrel outlet nozzle welds	Periodic enhanced visual (EVT-1) examination, no later than 2 refueling outages from the beginning of the license renewal period and subsequent examination on a ten-year interval.	100% of one side of the accessible surfaces of the selected weld and adjacent base metal (Note 4). See Figure A-4.

Table C-1 (cont.) MRP-227-A Primary Inspection and Monitoring Recommendations for Westinghouse-Designed Internals

Item	Applicability	Effect (Mechanism)	Expansion Link (Note 1)	Examination Method/Frequency (Note 1)	Examination Coverage
Core Barrel Assembly Upper and lower core barrel cylinder girth welds	All plants	Cracking (SCC, IASCC, Fatigue)	Upper and lower core barrel cylinder axial welds	Periodic enhanced visual (EVT-1) examination, no later than 2 refueling outages from the beginning of the license renewal period and subsequent examinations on a ten-year interval.	100% of one side of the accessible surfaces of the selected weld and adjacent base metal (Note 4). See Figure A-4
Core Barrel Assembly Lower core barrel flange weld (Note 5)	All plants	Cracking (SCC, Fatigue)	None	Periodic enhanced visual (EVT-1) examination, no later than 2 refueling outages from the beginning of the license renewal period and subsequent examinations on a ten-year interval.	100% of one side of the accessible surfaces of the selected weld and adjacent base metal (Note 4).
Baffle-Former Assembly Baffle-edge bolts	All plants with baffle-edge bolts NOTE: Applicable to SGS Unit 2	Cracking (IASCC, Fatigue) that results in • Lost or broken locking devices • Failed or missing bolts • Protrusion of bolt heads Aging Management (IE and ISR) (Note 6)	None	Visual (VT-3) examination, with baseline examination between 20 and 40 EFPY and subsequent examinations on a ten-year interval.	Bolts and locking devices on high fluence seams. 100% of components accessible from core side (Note 3). See Figures A-5 and A-7.

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Table C-1 (cont.) MRP-227-A Primary Inspection and Monitoring Recommendations for Westinghouse-Designed Internals

Item	Applicability	Effect (Mechanism)	Expansion Link (Note 1)	Examination Method/Frequency (Note 1)	Examination Coverage
Baffle-Former Assembly Baffle-former bolts	All plants	Cracking (IASCC, Fatigue) Aging Management (IE and ISR) (Note 6)	Lower support column bolts, Barrel-former bolts	Baseline volumetric (UT) examination between 25 and 35 EFPY, with subsequent examinations on a ten-year interval. (Note 8)	100% of accessible bolts (Note 3). Heads accessible from the core side. UT accessibility may be affected by complexity of head and locking device designs. See Figures A-5 and A-7.

Table C-1 (cont.) MRP-227-A Primary Inspection and Monitoring Recommendations for Westinghouse-Designed Internals

Item	Applicability	Effect (Mechanism)	Expansion Link (Note 1)	Examination Method/Frequency (Note 1)	Examination Coverage
Baffle-Former Assembly Assembly (Includes: Baffle plates, baffle edge bolts and indirect effects of void swelling in former plates)	All plants	Distortion (Void Swelling), or Cracking (IASCC) that results in: • Abnormal interaction with fuel assemblies • Gaps along high fluence baffle joint • Vertical displacement of baffle plates near high fluence joint • Broken or damaged edge bolt locking systems along high fluence baffle joint	None	Visual (VT-3) examination to check for evidence of distortion, with baseline examination between 20 and 40 EFPY and subsequent examinations on a ten-year interval.	Core side surface as indicated. See Figures A-5 and A-8.

Table C-1 (cont.) MRP-227-A Primary Inspection and Monitoring Recommendations for Westinghouse-Designed Internals

Item	Applicability	Effect (Mechanism)	Expansion Link (Note 1)	Examination Method/Frequency (Note 1)	Examination Coverage
Alignment and Interfacing Components Internals hold down spring	All plants with 304 stainless steel hold down springs NOTE: SGS Unit 2 hold down spring is 304 SS	Distortion (Loss of Load) Note: This mechanism was not strictly identified in the original list of age-related degradation mechanisms.	None	Direct measurement of spring height within three cycles of the beginning of the license renewal period. If the first set of measurements is not sufficient to determine life, spring height measurements must be taken during the next two outages, in order to extrapolate the expected spring height to 60 years.	Measurements should be taken at several points around the circumference of the spring, with a statistically adequate number of measurements at each point to minimize uncertainty. See Figure A-9.
Thermal Shield Assembly Thermal shield flexures	All plants with thermal shields NOTE: Applicable to SGS Unit 2	Cracking (Fatigue) or Loss of Material (Wear) that results in thermal shield flexures excessive wear, fracture, or complete separation	None	Visual (VT-3) no later than 2 refueling outages from the beginning of the license renewal period. Subsequent examinations on a ten-year interval.	100% of thermal shield flexures. See Figures A-4 and A-10.

Table C-1 (cont.) MRP-227-A Primary Inspection and Monitoring Recommendations for Westinghouse-Designed Internals

Item	Applicability	Effect (Mechanism)	Expansion Link (Note 1)	Examination Method/Frequency (Note 1)	Examination Coverage

Notes:

1. Examination acceptance criteria and expansion criteria for the Westinghouse components are in Table C-4.
2. A minimum of 75% of the total identified sample population must be examined.
3. A minimum of 75% of the total population (examined + unexamined), including coverage consistent with the Expansion criteria in C-4, must be examined for inspection credit.
4. A minimum of 75% of the total weld length (examined + unexamined), including coverage consistent with the Expansion criteria in C-4, must be examined from either the inner or outer diameter for inspection credit.
5. The lower core barrel flange weld may be alternatively designated as the core barrel-to-support plate weld in some Westinghouse plant designs.
6. Void swelling effects on this component is managed through management of void swelling on the entire baffle-former assembly.
7. WCAP-17451-P Revision 1 [44] requires a remote visual examination consistent with visual (VT-3) for minimum compliance and examination coverage of a minimum of 20% of the number of CRGT guide card assemblies. The baseline examination schedule has been adjusted for various CRGT designs, the extent of individual CRGT examination modified, and flexible subsequent examination regimens correlating to initial baseline sample size, accuracy of wear estimation and examination results. The content of this note and the noted cells is included according to the interim guidance provided in the EPRI transmittal MRP 2014-006 [48].
8. Baffle-former bolt inspection frequency is 10 years following the initial or baseline inspection, unless SGS Unit 2 provides an evaluation for NRC staff approval that justifies a longer interval between inspections.

Table C-2 MRP-227-A Expansion Inspection and Monitoring Recommendations for Westinghouse-Designed Internals

Item	Applicability	Effect (Mechanism)	Primary Link (Note 1)	Examination Method/Frequency (Note 1, Note 3)	Examination Coverage
Upper Internals Assembly Upper Core Plate	All plants	Cracking (Fatigue, Wear)	CRGT lower flange weld	Enhanced visual (EVT-1) examination. Re-inspection every 10 years following initial inspection.	100% of accessible surfaces (Note 2).
Lower Internals Assembly Lower support forging or castings	All plants NOTE: SGS Unit 2 has a lower support casting	Cracking Aging Management (TE in Casting)	CRGT lower flange weld	Enhanced visual (EVT-1) examination. Re-inspection every 10 years following initial inspection.	100% of accessible surfaces (Note 2). See Figure A-12.
Core Barrel Assembly Barrel-former bolts	All plants	Cracking (IASCC, Fatigue) Aging Management (IE, Void Swelling and ISR)	Baffle-former bolts	Volumetric (UT) examination Re-inspection every 10 years following initial inspection.	100% of accessible bolts. Accessibility may be limited by presence of thermal shields or neutron pads (Note 2). See Figure A-7.
Lower Support Assembly Lower support column bolts	All plants	Cracking (IASCC, Fatigue) Aging Management (IE and ISR)	Baffle-former bolts	Volumetric (UT) examination. Re-inspection every 10 years following initial inspection.	100% of accessible bolts or as supported by plant-specific justification (Note 2). See Figures A-11and A-12.

Table C-2 MRP-227-A Expansion Inspection and Monitoring Recommendations for Westinghouse-Designed Internals (cont.)

Item	Applicability	Effect (Mechanism)	Primary Link (Note 1)	Examination Method/Frequency (Note 1, Note 3)	Examination Coverage
Core Barrel Assembly Core barrel outlet nozzle welds	All plants	Cracking (SCC, Fatigue) Aging Management (IE of lower sections)	Upper core barrel flange weld	Enhanced visual (EVT-1) examination. Re-inspection every 10 years following initial inspection.	100% of one side of the accessible surfaces of the selected weld and adjacent base metal (Note 2). See Figure A-4.
Core Barrel Assembly Upper and lower core barrel cylinder axial welds	All plants	Cracking (SCC, IASCC) Aging Management (IE)	Upper and lower core barrel cylinder girth welds	Enhanced visual (EVT-1) examination. Re-inspection every 10 years following initial inspection.	100% of one side of the accessible surfaces of the selected weld and adjacent base metal (Note 2). See Figure A-4.
Lower Support Assembly Lower support column bodies (non cast)	All plants	Cracking (IASCC) Aging Management (IE)	Upper core barrel flange weld	Enhanced visual (EVT-1) examination. Re-inspection every 10 years following initial inspection.	100% of accessible surfaces (Note 2). See Figures A-12 and A-13.
Lower Support Assembly Lower support column bodies (cast)	All plants	Cracking (IASCC) including the detection of fractured support columns Aging Management (IE)	Control rod guide tube (CRGT) lower flanges	Visual (EVT-1) examination. Re-inspection every 10 years following initial inspection.	100% of accessible support columns (Note 2). See Figures A-12 and A-13.

Table C-2 MRP-227-A Expansion Inspection and Monitoring Recommendations for Westinghouse-Designed Internals (cont.)

Item	Applicability	Effect (Mechanism)	Primary Link (Note 1)	Examination Method/Frequency (Note 1, Note 3)	Examination Coverage
Bottom-Mounted Instrumentation System Bottom-mounted instrumentation (BMI) column bodies	All plants	Cracking (Fatigue) including the detection of completely fractured column bodies Aging Management (IE)	Control rod guide tube (CRGT) lower flanges	Visual (VT-3) examination of BMI column bodies as indicated by difficulty of insertion/withdrawal of flux thimbles. Re-inspection every 10 years following initial inspection. Flux thimble insertion/withdrawal to be monitored at each inspection interval.	100% of BMI column bodies for which difficulty is detected during flux thimble insertion/withdrawal. See Figures A-12 and A-14.

Notes:
1. Examination acceptance criteria and expansion criteria for the Westinghouse components are in Table C-4.
2. A minimum of 75% coverage of the entire examination area or volume, or a minimum sample size of 75% of the total population of like components of the examination is required (including both the accessible and inaccessible portions).
3. Expansion component inspection frequency is 10 years following the initial or baseline inspection, unless SGS Unit 2 provides an evaluation for NRC staff approval that justifies a longer interval between inspections.

Table C-3 MRP-227-A Existing Inspection and Aging Management Programs Credited in Recommendations for Westinghouse-Designed Internals

Item	Applicability	Effect (Mechanism)	Primary Link	Examination Method	Examination Coverage
Core Barrel Assembly Core barrel flange	All plants	Loss of material (Wear)	ASME Code Section XI	Visual (VT-3) examination to determine general condition for excessive wear.	All accessible surfaces at specified frequency.
Upper Internals Assembly Upper support ring or skirt	All plants	Cracking (SCC, Fatigue)	ASME Code Section XI	Visual (VT-3) examination.	All accessible surfaces at specified frequency.
Lower Internals Assembly Lower core plate XL lower core plate (Note 1)	All plants	Cracking (IASCC, Fatigue) Aging Management (IE)	ASME Code Section XI	Visual (VT-3) examination of the lower core plates to detect evidence of distortion and/or loss of bolt integrity.	All accessible surfaces at specified frequency.
Lower Internals Assembly Lower core plate XL lower core plate (Note 1)	All plants	Loss of material (Wear)	ASME Code Section XI	Visual (VT-3) examination.	All accessible surfaces at specified frequency.
Bottom-Mounted Instrumentation System Flux thimble tubes	All plants	Loss of material (Wear)	NUREG-1801, Rev. 1	Surface (ET) examination.	Eddy current surface examination, as defined in plant response to IEB 88-09.
Alignment and Interfacing Components Clevis insert bolts	All plants	Loss of material (Wear) (Note 2)	ASME Code Section XI	Visual (VT-3) examination.	All accessible surfaces at specified frequency.

Table C-3 (cont.) MRP-227-A Existing Inspection and Aging Management Programs Credited in Recommendations for Westinghouse-Designed Internals

Item	Applicability	Effect (Mechanism)	Primary Link	Examination Method	Examination Coverage
Alignment and Interfacing Components Upper core plate alignment pins	All plants	Loss of material (Wear)	ASME Code Section XI	Visual (VT-3) examination.	All accessible surfaces at specified frequency.

Notes:
1. XL = "Extra Long," referring to Westinghouse plants with 14-foot cores.
2. Bolt was screened in because of stress relaxation and associated cracking; however, wear of the clevis/insert is the issue.

Table C-4 MRP-227-A Acceptance Criteria and Expansion Criteria Recommendations for Westinghouse-Designed Internals

Item	Applicability	Examination Acceptance Criteria (Note 1)	Expansion Link(s)	Expansion Criteria	Additional Examination Acceptance Criteria
Control Rod Guide Tube Assembly Guide plates (cards)	All plants	Visual (VT-3) examination. (Note 3) The specific relevant condition is wear that could lead to loss of control rod alignment and impede control assembly insertion.	None	N/A	N/A

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Table C-4 (cont.) MRP-227-A Acceptance Criteria and Expansion Criteria Recommendations for Westinghouse-Designed Internals

Item	Applicability	Examination Acceptance Criteria (Note 1)	Expansion Link(s)	Expansion Criteria	Additional Examination Acceptance Criteria
Control Rod Guide Tube Assembly Lower flange welds	All plants	Enhanced visual (EVT-1) examination. The specific relevant condition is a detectable crack-like surface indication.	a. Bottom-mounted instrumentation (BMI) column bodies b. Lower support column bodies (cast), upper core plate and lower support forging or casting	a. Confirmation of surface-breaking indications in two or more CRGT lower flange welds, combined with flux thimble insertion/withdrawal difficulty, shall require visual (VT-3) examination of BMI column bodies by the completion of the next refueling outage. b. Confirmation of surface-breaking indications in two or more CRGT lower flange welds shall require EVT-1 examination of cast lower support column bodies, upper core plate and lower support forging/castings within three fuel cycles following the initial observation.	a. For BMI column bodies, the specific relevant condition for the VT-3 examination is completely fractured column bodies. b. For cast lower support column bodies, upper core plate and lower support forging/castings, the specific relevant condition is a detectable crack-like surface indication.

Table C-4 (cont.) MRP-227-A Acceptance Criteria and Expansion Criteria Recommendations for Westinghouse-Designed Internals

Item	Applicability	Examination Acceptance Criteria (Note 1)	Expansion Link(s)	Expansion Criteria	Additional Examination Acceptance Criteria
Core Barrel Assembly Upper core barrel flange weld	All plants	Periodic enhanced visual (EVT-1) examination. The specific relevant condition is a detectable crack-like surface indication.	a. Core barrel outlet nozzle welds b. Lower support column bodies (non cast)	a. The confirmed detection and sizing of a surface-breaking indication with a length greater than two inches in the upper core barrel flange weld shall require that the EVT-1 examination be expanded to include the core outlet nozzle welds by the completion of the next refueling outage. b. If extensive cracking in the core barrel outlet nozzle welds is detected, EVT-1 examination shall be expanded to include the upper six inches of the accessible surfaces of the non-cast lower support column bodies within three fuel cycles follow the initial observation.	a and b. The specific relevant condition for the expansion core barrel outlet nozzle weld and lower support column body examination is a detectable crack-like surface indication.

Table C-4 (cont.) MRP-227-A Acceptance Criteria and Expansion Criteria Recommendations for Westinghouse-Designed Internals

Item	Applicability	Examination Acceptance Criteria (Note 1)	Expansion Link(s)	Expansion Criteria	Additional Examination Acceptance Criteria
Core Barrel Assembly Lower core barrel flange weld (Note 2)	All plants	Periodic enhanced visual (EVT-1) examination. The specific relevant condition is a detectable crack-like surface indication.	None	None	None
Core Barrel Assembly Upper core barrel cylinder girth welds	All plants	Periodic enhanced visual (EVT-1) examination. The specific relevant condition is a detectable crack-like surface indication.	Upper core barrel cylinder axial welds	The confirmed detection and sizing of a surface-breaking indication with a length greater than two inches in the upper core barrel cylinder girth welds shall require that the EVT-1 examination be expanded to include the upper core barrel cylinder axial welds by the completion of the next refueling outage.	The specific relevant condition for the expansion upper core barrel cylinder axial weld examination is a detectable crack-like surface indication.

Table C-4 (cont.) MRP-227-A Acceptance Criteria and Expansion Criteria Recommendations for Westinghouse-Designed Internals

Item	Applicability	Examination Acceptance Criteria (Note 1)	Expansion Link(s)	Expansion Criteria	Additional Examination Acceptance Criteria
Core Barrel Assembly Lower core barrel cylinder girth welds	All plants	Periodic enhanced visual (EVT-1) examination. The specific relevant condition is a detectable crack-like surface indication.	Lower core barrel cylinder axial welds	The confirmed detection and sizing of a surface-breaking indication with a length greater than two inches in the lower core barrel cylinder girth welds shall require that the EVT-1 examination be expanded to include the lower core barrel cylinder axial welds by the completion of the next refueling outage.	The specific relevant condition for the expansion lower core barrel cylinder axial weld examination is a detectable crack-like surface indication.
Baffle-Former Assembly Baffle-edge bolts	All plants with baffle-edge bolts NOTE: Applicable to SGS Unit 2	Visual (VT-3) examination. The specific relevant conditions are missing or broken locking devices, failed or missing bolts, and protrusion of bolt heads.	None	N/A	N/A

Table C-4 (cont.) MRP-227-A Acceptance Criteria and Expansion Criteria Recommendations for Westinghouse-Designed Internals

Item	Applicability	Examination Acceptance Criteria (Note 1)	Expansion Link(s)	Expansion Criteria	Additional Examination Acceptance Criteria
Baffle-Former Assembly Baffle-former bolts	All plants	Volumetric (UT) examination. The examination acceptance criteria for the UT of the baffle-former bolts shall be established as part of the examination technical justification.	a. Lower support column bolts b. Barrel-former bolts	a. Confirmation that more than 5% of the baffle-former bolts actually examined on the four baffle plates at the largest distance from the core (presumed to be the lowest dose locations) contain unacceptable indications shall require UT examination of the lower support column bolts within the next three fuel cycles. b. Confirmation that more than 5% of the lower support column bolts actually examined contain unacceptable indications shall require UT examination of the barrel-former bolts.	a and b. The examination acceptance criteria for the UT of the lower support column bolts and the barrel-former bolts shall be established as part of the examination technical justification.

Table C-4 (cont.) MRP-227-A Acceptance Criteria and Expansion Criteria Recommendations for Westinghouse-Designed Internals

Item	Applicability	Examination Acceptance Criteria (Note 1)	Expansion Link(s)	Expansion Criteria	Additional Examination Acceptance Criteria
Baffle-Former Assembly Assembly	All plants	Visual (VT-3) examination. The specific relevant conditions are evidence of abnormal interaction with fuel assemblies, gaps along high fluence shroud plate joints, vertical displacement of shroud plates near high fluence joints, and broken or damaged edge bolt locking systems along high fluence baffle plate joints.	None	N/A	N/A

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Table C-4 (cont.) MRP-227-A Acceptance Criteria and Expansion Criteria Recommendations for Westinghouse-Designed Internals

Item	Applicability	Examination Acceptance Criteria (Note 1)	Expansion Link(s)	Expansion Criteria	Additional Examination Acceptance Criteria
Alignment and Interfacing Components Internals hold down spring	All plants with 304 stainless steel hold down springs NOTE: SGS Unit 2 hold down spring is 304 SS	Direct physical measurement or spring height. The examination acceptance criterion for this measurement is that the remaining compressible height of the spring shall provide hold-down forces within the plant-specific design tolerance.	None	N/A	N/A

Table C-4 (cont.) MRP-227-A Acceptance Criteria and Expansion Criteria Recommendations for Westinghouse-Designed Internals

Item	Applicability	Examination Acceptance Criteria (Note 1)	Expansion Link(s)	Expansion Criteria	Additional Examination Acceptance Criteria
Thermal Shield Assembly Thermal shield flexures	All plants with thermal shields NOTE: Applicable to SGS Unit 2	Visual (VT-3) examination. The specific relevant conditions for thermal shield flexures are excessive wear, fracture, or complete separation.	None	N/A	N/A

Notes:
1. The examination acceptance criterion for visual examination is the absence of the specified relevance condition(s).
2. The lower core barrel flange weld may alternatively be designated as the core barrel-to-support plate weld in some Westinghouse plant designs.
3. WCAP-17451-P Revision 1 [44] specifies a remote visual examination consistent with visual (VT-3) but allows for various supplemental measurement techniques which if employed increase wear estimate accuracy and allow use of acceptance criteria (wear projections) to determine the appropriate re-examination interval. The content of this note and the noted cell is included according to the interim guidance provided in the EPRI transmittal MRP 2014-006 [48].