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Program Plan for Aging Management of Reactor Vessel Internals at Diablo Canyon Power Plant Unit 1

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LIST OF ACRONYMS

3-D	three-dimensional
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
CRGT	control rod guide tube
CUF	cumulative usage factor
DCPP	Diablo Canyon Power Plant
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 mode, effects, and criticality analysis
GALL	Generic Aging Lessons Learned
I&E	inspection and evaluation
IASCC	irradiation-assisted stress corrosion cracking
IE	irradiation embrittlement
IGSCC	intergranular stress corrosion cracking
INPO	Institute of Nuclear Power Operations
IP	industry program
ISI	inservice inspection
ISR	irradiation-enhanced stress relaxation
LRA	License Renewal Application
MRP	Materials Reliability Program
NDE	nondestructive examination
NEI	Nuclear Energy Institute
NOS	Nuclear Oversight Section
NRC	United States Nuclear Regulatory Commission
NSSS	nuclear steam supply system
OE	operating experience
OER	Operating Experience Report
PH	precipitation-hardenable
PG&E	Pacific Gas and Electric Company
PWR	pressurized water reactor
PWROG	Pressurized Water Reactor Owners Group (formerly WOG)

LIST OF ACRONYMS (cont.)

PWSCC	primary water stress corrosion cracking
QA	quality assurance
RCS	reactor coolant system
RFO	refueling outage
RV	reactor vessel
RVI	reactor vessel internals
SCC	stress corrosion cracking
SE	safety evaluation
SER	Safety Evaluation Report
SRP	Standard Review Plan
SS	stainless steel
TE	thermal embrittlement
UFSAR	Updated Final Safety Analysis Report
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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1 PURPOSE

The purpose of this report is to document the Diablo Canyon Power Plant (DCPP) Unit 1, hereafter DCPP Unit 1, Reactor Vessel Internals (RVI) Aging Management Program Plan (AMP). The purpose of the AMP is to manage the effects of aging on reactor vessel internals through the license renewal period, which for DCPP Unit 1 begins at midnight on November 2, 2024. This document provides a description of the program as it relates to the management of aging effects identified in various regulatory, industry-generated, and DCPP Unit 1 plant-specific documents in support of license renewal program evaluations. This AMP is prepared in accordance with various regulatory and industry-generated documents, and is supported by existing DCPP Unit 1 documents and procedures. As required, by industry experience or directive in the future, the DCPP Unit 1 RVI AMP will be updated or supported by additional documents to provide clear and concise direction for the effective management of aging degradation in the DCPP Unit 1 reactor internals components. These actions provide assurance that operations at DCPP Unit 1 will continue to be conducted in accordance with the current licensing basis (CLB) for the reactor vessel internals by fulfilling license renewal commitments (Reference 1), American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel (B&PV) Code Section XI Inservice Inspection (ISI) requirements (Reference 2), and industry requirements (Reference 3). 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 DCPP Unit 1 RVI AMP are to:

- Demonstrate that the effects of aging on the RVI will be adequately managed for the period of extended operation in accordance with Title 10 of the Code of Federal Regulations (CFR) Part 54 (Reference 4).
- Summarize the role of existing DCPP Unit 1 AMPs in the RVI AMP.
- Define and implement the industry-defined (EPRI/MRP and PWROG) pressurized water reactor (PWR) RVI requirements and guidance for managing aging of reactor internals.
- Provide an inspection plan summary for the DCPP Unit 1 reactor internals.

DCPP License Renewal Commitments 72 & 73 (Reference 34) defines the content and timeline for the program that Pacific Gas and Electric Company (PG&E) has committed to implement for the reactor vessel internals components:

> *Commitment 72*: Prior to the period of extended operation, implement the PWR Vessel Internals Program to conform to LR-ISG-2011-04 as discussed in PG&E Letter DCL-14-103, Enclosure 1, Attachment 4, including the plant-specific action items, conditions, and limitations identified in the U.S. Nuclear Regulatory Commission (NRC) Safety Evaluation (SE) for MRP-227.
>
> *Commitment 73*: The NRC SE for MRP-227 contains eight action items for applicants/licensees to consider. Responses to the applicable aging management program plant-specific action items,

conditions, and limitations identified in the NRC SE on MRP-227 will be submitted to the NRC by December 2015. Reference DCP-14-103, Enclosure 1, Attachment 4.

Augmented inspections, based on required program enhancements resulting from ongoing and future industry programs, will be incorporated into the DCPP Unit 1 RVI AMP. The Program Plan for Inspection of Reactor Vessel Internals will coordinate with the existing DCPP Unit 1 Inservice Inspection Program (Reference 2), and will supplement that program with the augmented examinations for managing the potential aging effects of the reactor vessel internals. Corrective actions for augmented inspections will either be developed using repair and replacement procedures equivalent to those requirements in ASME B&PV Code, Section XI (Reference 5), or equivalent or more rigorous procedures will be determined by PG&E independently or in cooperation with the industry. PG&E is currently committed to the 2001 Edition through the 2003 Addenda of the ASME Code for DCPP Unit 1, and initial development of this AMP will be based on this edition; however, for future development, later editions and addenda will be invoked as required by 10 CFR 50.55a or approved NRC Code Cases or Safety Evaluation Reports (SERs).

This AMP for the DCPP Unit 1 reactor internals demonstrates that the Reactor Vessel Internals Program adequately manages the effects of aging for reactor internals components. The AMP also establishes the basis for providing reasonable assurance that the internals components will continue to perform their intended function through the DCPP Unit 1 license renewal period of extended operation. It also supports the DCPP Unit 1 License Renewal Commitment 72 to implement the PWR Vessel Internals Program to conform to LR-ISG-2011-04 as discussed in PG&E Letter DCL-14-103, Enclosure 1, Attachment 4, including the plant-specific action items, conditions, and limitations identified in the NRC Safety Evaluation, Revision 1, for MRP-227. This AMP also supports the DCPP Unit 1 License Renewal Commitment 73, which is to provide responses to the applicable aging management program plant-specific action items, conditions, and limitations identified in the NRC SE, Revision 1, on MRP-227 to the NRC by December 2015. Furthermore, this AMP will demonstrate the consistency of the program with that documented in NUREG-1801, December 2010, Section XI.M16A (Reference 6). The development of this program satisfies the MRP-227 requirements.

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) (Reference 7). In recent years, the U.S. nuclear power industry has been actively engaged 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 (WOG) issued WCAP-14577 (Reference 8), "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, the EPRI MRP engaged in an effort 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, and on the accumulated industry research data, the following elements of an Aging Management Program were further developed (References 8, 9 and 10):

- 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. These categories ranged from components for which the effects from the postulated aging mechanisms are insignificant, to components that are moderately susceptible to the aging effects, to 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 for implementing the appropriate aging management methodology, baseline examination timing, and the need and timing of subsequent inspections were developed. Development of these strategies was based on combining the results of functionality assessment with several contributing factors, including component accessibility, operating experience, existing evaluations, and prior examination results.

The industry effort, as coordinated by the EPRI MRP, has finalized initial Inspection and Evaluation (I&E) Guidelines for reactor internals and submitted the document to the NRC with a request for a formal SER. A supporting document addressing inspection requirements has also been completed. The industry guidance is contained within two separate EPRI MRP documents:

- MRP-227-A, "PWR Internals Inspection and Evaluation Guidelines" (Reference 3) (hereafter referred to as "the I&E Guidelines" or simply "MRP-227-A") provides the industry background, listing of reactor internals components requiring inspection, type of non-destructive evaluation (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, "Inspection Standard for PWR Internals" (Reference 11) provides guidance on the qualification and demonstration of the NDE techniques and on other criteria pertaining to the actual performance of the inspections.

The PWROG has also developed "Reactor Internals Acceptance Criteria Methodology and Data Requirements" for the MRP-227 inspections, where feasible (Reference 12). Final reports are 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 if a generic approach is not practical.

The DCPP reactor internals for Unit 1 are integral to the reactor coolant system (RCS) of a Westinghouse four-loop nuclear steam supply system (NSSS). A typical illustration of the reactor vessel internals is provided in Figure A-1.

The DCPP License Renewal Application (LRA), Section 2.3.1, (Reference 13) describes the DCPP Unit 1 reactor vessel internals. The RVI consist of the lower core support structure (including the entire core barrel and the thermal shield), the upper core support structure, and the incore instrumentation support structures. The reactor internals support the core, maintain fuel alignment, limit fuel assembly movement, maintain alignment between fuel assemblies and CRDMs, direct coolant flow to the pressure vessel head, provide gamma and neutron shielding, and provide guides for incore instrumentation.

The lower core support structure includes the baffle and former plates, core barrel assembly, thermal shield, lower core plates, core support casting, support columns, secondary core support, energy absorbers, tie plates, manway cover, and support ring.

The upper core support structure includes the upper support columns, upper support plate, upper core plate, and control rod guide tubes.

The incore instrumentation support structure includes the flux thimble tubes and guide tubes, seal table and fittings, and upper instrumentation columns. Components that provide interfaces between the major assemblies include the radial keys, clevis inserts, fuel alignment pins, head/vessel alignment pins, upper core plate alignment pins, and hold-down spring.

The NRC has issued an SER to Diablo Canyon that states that the NRC will not finalize a decision on license renewal until completion of three-dimensional (3-D) seismic studies and PG&E's receipt of a coastal consistency certification. The staff will supplement this SER, as necessary, considering any relevant new information from the seismic studies, operating experience, and annual updates prior to finalizing a decision on license renewal. In the SER, the NRC concluded that the DCPP LRA (Reference 13) adequately identified the RVI systems, structures, and components that are subject to an aging management review (AMR), and that the requirements of 10 CFR 54.29(a) (Reference 4) had been met. Appendix B, Table B-1 of this report lists the DCPP Unit 1 reactor vessel internals components and subcomponents subject to AMR requirements according to Table 3.1.2-1 of the DCPP LRA.

The U.S. nuclear industry, as noted through the efforts of the MRP and PWROG, has further investigated the components and subcomponents that require aging management to support continued reliable functioning. As designated by the Nuclear Energy Institute (NEI) protocol NEI 03-08, "Guidelines for the Management of Materials Issues" (Reference 14), 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. MRP-227 was issued in December 2008, and plant AMPs must have been completed by December 2011 (or sooner) if required by plant-specific license renewal commitments. Per the MRP-227 requirement, DCPP Unit 1 completed development of the AMP by the due date of December 2011 via issuance of Revision 0 of this WCAP.

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 in a manner consistent with the current licensing basis for the period of extended operation.

3 PROGRAM OWNER

The successful implementation and comprehensive long-term management of the DCPP Unit 1 RVI AMP will require PG&E to interact with multiple industry organizations including, but not limited to, the ASME, MRP, NRC, and PWROG. The responsibilities of the individual PG&E organizational groups are provided in the following paragraphs. PG&E will maintain cognizance of industry activities related to PWR internals inspection and aging management and will address and implement industry guidance stemming from those activities, as appropriate under NEI 03-08 practices.

The overall responsibility for administration of the RVI AMP lies with PG&E Senior Management.

Additional responsibilities and the appropriate responsible personnel are discussed in the following subsections.

3.1 ENGINEERING DIRECTOR

- Approves implementation of the RVI AMP.
- Ensures coordination and implementation of the RVI AMP.

3.2 RCS MATERIALS DEGRADATION MANAGEMENT PROGRAM OWNER

- Acts as the point of contact with materials-related industry programs (IPs) and NEI
- Ensures new requirements or recommendations issued under NEI 03-08 are disseminated to the appropriate program owners or responsible personnel.
- Initiates Notifications in the Corrective Action Program for tracking of NEI 03-08 Mandatory and Needed requirements.
- Ensures timely reporting and management of new or unexpected RCS material issues in accordance with the emergent issue protocol of NEI 03-08.
- Processes any deviations taken from IP guidelines in accordance with NEI 03-08 requirements.

3.3 REACTOR VESSEL INTERNALS PROGRAM MANAGER

- Overall development of the RVI AMP.
- Administers and oversees implementation of the RVI AMP.
- Ensures that regulatory requirements related to inspection activities are met and incorporated into the RVI AMP.
- Communicates with senior management on relevant industry experience.

- Maintains the RVI AMP to incorporate changes and updates based on industry operating experience and benchmarking results.

- Ensures prompt notification of the RCS Materials Degradation Management Program Owner whenever an issue or indication of potential generic Industry significance is identified.

- Participates in the planning and implementation of inspections of the RVI.

- Participates in industry programs related to RVI AMP.

3.4 INSERVICE INSPECTION ENGINEER

- Plans and implements inspections required by ASME Section XI B-N-3, the supplemental inspections identified in this RVI AMP, and any other plant-specific commitments for inspections related to managing the aging of RVI.

- Reviews, evaluates, and dispositions inspection results.

- Reviews and approves vendor NDE procedures and personnel qualifications.

- Provides direction and oversight of contracted NDE activities.

3.5 WORK CONTROL MANAGER

- Integrates required activities into the appropriate outage plans.

3.6 CHEMISTRY MANAGER

- Maintains primary water chemistry in accordance with approved DCPP procedures and specifications.

- Ensures that DCPP documentation supports and incorporates the guidance of industry programs including, but not limited to, the EPRI Primary Water Chemistry Guidelines (Reference 9).

4 DESCRIPTION OF THE DIABLO CANYON POWER PLANT UNIT 1 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 aging management programs based on inspection and evaluation. The intent of the DCPP Unit 1 AMP is to ensure the long-term integrity and safe operation of the reactor internals components. PG&E has developed this AMP in conformance with the 10 Generic Aging Lessons Learned (GALL) (Reference 6) attributes and MRP-227-A (Reference 3).

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 (References 9 and 15), inspections prescribed by the "Inservice Inspection Program Implementation" (Reference 2), thimble tube inspections (Reference 18), and past and future mitigation projects such as control rod guide tube support pin replacement, combined with augmented inspections or evaluations as recommended by MRP-227-A.

Aging degradation mechanisms that affect internals have been identified for DCPP Unit 1 and are documented in the LRA submitted for DCPP (Reference 13). 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 DCPP Unit 1 AMR methodology 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. The actual mechanism that causes SCC involves a complex interaction of environmental and metallurgical factors. The aging effect is cracking.

- <u>Primary Water Stress Corrosion Cracking</u>

  Primary water stress corrosion cracking (PWSCC) is a unique form of SCC that occurs as a result of the chemistry of primary coolant acting on primary components fabricated from susceptible materials. 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.

- Wear (loss of material)

  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 (cracking)

  Fatigue is the structural deterioration that can occur as the result of repeated stress/strain cycles caused by fluctuating loads and/or 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 (reduction in fracture toughness)

  Thermal aging embrittlement is the exposure of delta ferrite within cast austenitic stainless steel (CASS) and precipitation-hardenable (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 (reduction in fracture toughness)

  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 (distortion)

  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 incore 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 (loss of preload, or loss of mechanical closure integrity)

  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 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 at PWR internals temperatures.

  Creep (or more precisely, secondary creep) is a slow, time- and temperature-dependent, plastic deformation of materials that can occur at 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 considering 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 DCPP 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 (Reference 6). In the DCPP Unit 1 RVI AMP, this is demonstrated through application of the existing DCPP Unit 1 AMR methodology that credits inspections prescribed by the ASME Section XI Inservice Inspection Program, which will become part of the DCPP Unit 1 Inspection Program Plan for Reactor Vessel Internals along with existing DCPP Unit 1 programs, and additional augmented inspections based on MRP-227-A recommendations. The Program Plan for Inspection of Reactor Vessel Internals will coordinate with the existing DCPP Unit 1 Inservice Inspection Program (Reference 2) and will supplement that program with the augmented examinations for managing the potential aging effects of the reactor vessel internals. A description of the applicable existing DCPP Unit 1 programs and compliance with the elements of the GALL is contained in the following subsections.

4.1 EXISTING DIABLO CANYON POWER PLANT UNIT 1 PROGRAMS

PG&E's overall strategy for managing aging in reactor internals components at DCPP Unit 1 is supported by the following existing programs:

- Water Chemistry
- ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD
- Flux Thimble Tube Inspection

These are established programs that support the aging management of RCS components in addition to the RVI components. Although affiliated with and supporting the RVI AMP, they will be managed under the existing programs.

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

4.1.1 Water Chemistry

The DCPP Unit 1 Primary Strategic Water Chemistry Plan (Reference 15) is used to mitigate aging effects on component surfaces that are exposed to PWR primary water as process fluid. Chemistry programs are used to control water chemistry for impurities that accelerate corrosion and contaminants that may cause cracking due to SCC. This program relies on monitoring and control of water chemistry to keep operating levels of various contaminants below the system-specific limits. The Primary Strategic Water Chemistry Plan is based on the EPRI PWR Primary Water Chemistry Guidelines (Reference 9). The limits imposed by the DCPP Unit 1 program meet the intent of the industry standard for addressing primary water chemistry (Reference 9).

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

4.1.2 ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD

The DCPP Unit 1 Inservice Inspection Program Implementation (Reference 2) is used to monitor for aging effects such as cracking, loss of preload due to stress relaxation or irradiation creep, loss of material, and reduction of fracture toughness due to thermal embrittlement. For DCPP Unit 1, inspections conducted under the reactor internals AMPs will be controlled as a combination of ASME Section XI ISI exams on core support structures and augmented exams performed under that ISI Program, which will become part of the DCPP Unit 1 Inspection Program Plan for Reactor Vessel Internals for the remaining reactor internals components addressed within MRP-227-A. The Program Plan for Inspection of Reactor Vessel Internals will coordinate with the existing DCPP Unit 1 Inservice Inspection Program (Reference 2) and will supplement that program with the augmented examinations for managing the potential aging effects of the reactor vessel internals. The DCPP Unit 1 Section XI, 10-year

ISI examination supporting the license renewal period is scheduled to take place during Spring 2024, Cycle 24. This is based on average 20-month cycles.

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

4.1.3 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 and 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. In turn, the flux thimbles provide paths for the neutron flux detectors into the core and are subject to reactor coolant pressure on the outside and containment pressure on the inside.

The DCPP Unit 1 Flux Thimble Tube Inspection program is an existing plant-specific program that satisfies NRC Bulletin 88-09 (Reference 19) requirements that a tube wear inspection procedure (References 17 and 18) be established and maintained for Westinghouse-supplied reactors that use bottom-mounted flux thimble tube instrumentation. Details of the program are given in the DCPP LRA, Section B2.1.21, page B-93. The program includes eddy current testing requirements for thimble tubes and criteria for determining sample size (includes all thimble tubes installed in the reactor vessel), inspection frequency, flaw evaluation, and corrective actions. The Flux Thimble Tube Inspection Program effectively manages aging effects by identifying loss of material due to wear in the thimble tubes prior to leakage. Continued implementation of this program provides reasonable assurance that aging effects will be managed such that the BMI thimble tubes will continue to perform their intended function consistent with the current licensing basis for the period of extended operation.

The SER for the DCPP LRA (Reference 1) reviewed the Flux Thimble Tube Inspection program against the 10 program elements of the GALL (Reference 6) for Program XI.M37 and determined that the aging effects would be adequately managed for the period of extended operation. The SER evaluation remains applicable.

4.2 SUPPORTING DIABLO CANYON POWER PLANT UNIT 1 PROGRAMS AND AGING MANAGEMENT SUPPORTIVE PLANT ENHANCEMENTS

4.2.1 Reactor Internals Aging Management Review Process

A comprehensive review of aging management was performed for the DCPP Unit 1 reactor vessel internals components according to the requirements of the License Renewal Rule (Reference 4). This review was conducted in support of the DCPP Unit 1 license renewal for reactor internals (Reference 33). The DCPP LRA (Reference 13) was approved by the NRC in Reference 1. Subsection 2.3.1.1 and Table 2.3.1-1 of the LRA identified the components that are subject to AMR. Table 3.1.2-1 of the LRA provides the detailed results of the AMRs conducted on these components and includes a comparison to NUREG-1801, Volume 2 to note consistencies. Appendix B, Table B-1 of the DCPP Unit 1 AMP includes a portion of LRA Table 3.1.2-1.

The aging management review supported the 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 to augment existing programs or to create new aging management programs if the existing programs were found to be inadequate to manage the aging effects

Aging management reviews were performed for each DCPP Unit 1 system that contained long-lived, passive components requiring aging management review, and the results are incorporated into the DCPP LRA.

4.2.2 Flux Thimble Tubes

The Flux Thimble Tube Inspection program manages loss of material by performing wall thickness eddy current testing of all flux thimble tubes that form part of the RCS pressure boundary. The pressure boundary includes the length of the tube inside the reactor vessel out to the seal fittings outside the reactor vessel. Eddy current testing is performed on the portion of the tubes inside the reactor vessel. The Flux Thimble Tube Inspection program does not prevent degradation due to aging effects but provides measures for inspection and evaluation to detect the degradation prior to loss of intended function. The program implements the recommendations of NRC Bulletin 88-09 (Reference 19).

All flux thimble tubes are currently inspected during each refueling outage. Wall thickness measurements are trended and wear rates are calculated. If the current measured wear exceeds the acceptance criteria or the predicted wear (as a measure of percent through wall), or if for a given flux thimble tube is projected to exceed the established acceptance criteria for wall thickness prior to the next refueling outage, corrective actions are taken to reposition, cap, or replace the tube. Program documentation maintains details regarding the core location, wear location, and the number of times a tube has been previously repositioned or replaced. Any thimble tube exhibiting an abnormally high wear rate is capped or replaced. Design changes are also implemented to use more wear-resistant thimble tube materials (e.g., chrome-plated stainless steel). The inspection frequency may be revised as appropriate based upon items such as operating experience and recommendations from the PWROG.

4.2.3 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. The limits imposed by the Primary Water Chemistry Plan (Reference 15) at DCPP Unit 1 are consistent with the EPRI Primary Water Chemistry Guidelines as described in Section 4.1.

The original DCPP Unit 1 support pins were fabricated from *INCONEL*® alloy X-750 that was hot rolled, solution treated or annealed, 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 SCC and likely to fail during the lifetime of a nuclear power plant. To address the susceptibility the support pins were replaced with cold-worked 316 stainless steel, with a design and stress distribution modified to be highly resistant to SCC.

Support pins were replaced at DCPP Unit 1 and detailed descriptions of the replacement are retained in the plant records (Reference 20).

4.2.4 Power Uprating Project

Unit 1 was designed to operate at up to a maximum power of 3488 MWt core power but was initially licensed for only 3338 MWt core power even though most safety-related analyses of record were performed assuming 3411 MWt core power. Therefore, a power uprate of Unit 1 was performed to increase the core power to 3411 MWt, which is consistent with the DCPP Unit 2 power level. Performance of this power uprate was approved by the NRC per Diablo Canyon Unit 1 License Amendment 143 and resulted in greater power generation of electricity and improved similarity between the units. The power uprate design evaluation (Reference 21) ensured the unit remained consistent with safety-related analyses and remained below design basis limits.

The evaluation of the power uprate performed at DCPP Unit 1 and detailed descriptions of the changes are detailed in plant records (Reference 21).

4.2.5 Reactor Vessel Internals Program

The PWR Vessel Internals Program is a new program that will be implemented prior to the period of extended operation. The program relies on implementation of the inspection and evaluation guidelines in MRP-227-A and MRP-228 to manage the aging effects of the reactor vessel internals components. This program is discussed in PG&E Letter DCL-14-103 (Reference 34) and is used to manage: (a) cracking, including stress corrosion cracking, primary water stress corrosion cracking, irradiation-assisted stress corrosion cracking, and cracking due to fatigue/cyclical loading; (b) loss of material induced by wear; (c) loss of fracture toughness due to either thermal aging embrittlement, irradiation embrittlement, or void swelling; (d) dimensional changes due to void swelling or distortion and loss of preload due to thermal and irradiation-enhanced stress relaxation or irradiation-enhanced creep.

4.3 INDUSTRY PROGRAMS

4.3.1 WCAP-14577, Aging Management for Reactor Internals

The WOG (now PWROG) topical report WCAP-14577 (Reference 8) contains a technical evaluation of aging degradation mechanisms and aging effects for Westinghouse RVI components. The LRA was completed using the interim methodology in WCAP-14577 (Reference 8). The WOG sent the report to the NRC staff to demonstrate that WOG member plant owners that subscribed to the WCAP could adequately manage effects of aging on RVI during the period of extended operation, using approved aging management methodologies of the WCAP to develop plant-specific aging management programs.

The aging management review for the DCPP Unit 1 internals was completed in accordance with the requirements of WCAP-14577 (Reference 8).

4.3.2 MRP-227, Reactor Internals Inspection and Evaluation Guidelines

MRP-227-A (Reference 3), 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.

4.3.2.1 MRP-227 RVI Component Categorizations

MRP-227 used a screening and ranking process to aid in the identification of required inspections for specific RVI components. MRP-227 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 United States 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 as "Primary" components, "Expansion" components, "Existing Programs" components, or "No Additional Measures" components, as 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 are not intended to supersede any ASME B&PV Code Section XI 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.

4.3.2.2 NEI 03-08 Guidance within MRP-227

The industry program requirements of MRP-227 are classified in accordance with the requirements of the NEI 03-08 (Reference 14) protocols. The MRP-227 guideline includes Mandatory, Needed, and Good Practice elements as follows:

- **Mandatory**

There is one Mandatory element:

> *Each commercial U.S. PWR unit shall develop and document a program for management of aging of reactor internals components within 36 months following issuance of MRP-227, Rev. 0 (that is, no later than December 31, 2011).*
>
> DCPP Unit 1 Applicability: MRP-227, Revision 0, was officially issued by the industry in December 2008. An AMP was therefore required to be developed by December 2011. PG&E satisfied this requirement via issuance of Revision 0 of this WCAP in December 2011.

- **Needed**

There are five needed elements:

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.*

   DCPP Unit 1 Applicability: MRP-227-A augmented inspections have been appropriately incorporated into the DCPP RVI AMP program procedure, TS1.ID11 (Reference 43). The Program Plan for Inspection of Reactor Vessel Internals will coordinate with the existing DCPP Unit 1 Inservice Inspection Program (Reference 2) and will supplement that program with the augmented examinations for managing the potential aging effects of the reactor vessel internals. The applicable Westinghouse tables contained in MRP-227-A are Table 4-3 (Primary), Table 4-6 (Expansion), Table 4-9 (Existing), and Table 5-3 (Acceptance Criteria and Expansion Criteria

Recommendations) 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 Inspection Standard [MRP-228].*

   DCPP Unit 1 Applicability: Inspection standards will be in accordance with the requirements of MRP-228 (Reference 11). These inspection standards will be used for augmented inspection at DCPP 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.*

   DCPP Unit 1 Applicability: PG&E 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.*

   DCPP Unit 1 Applicability: As discussed in subsection 4.3.3, PG&E 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 an NRC-approved evaluation methodology.*

   DCPP Unit 1 Applicability: PG&E 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.

4.3.2.3 GALL AMP Development Guidance

It should be noted that MRP-227-A, Appendix A (Reference 3) also includes a description of the attributes that make up an acceptable AMP. These attributes are similar to the previously discussed attributes of Revision 2 of the GALL Report and are consistent with the PG&E Aging Management Review process. Evaluation of the DCPP Unit 1 RVI AMP against GALL attribute elements is provided in Section 5 of this program plan.

As part of License Renewal, PG&E agreed to participate in industry activities associated with the development of the standard Industry Guidelines 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 DCPP Unit 1 RVI AMP.

4.3.2.4 MRP-227-A Applicability to Diablo Canyon Power Plant Unit 1

The applicability of MRP-227-A to DCPP 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.*

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

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

  DCPP Unit 1 Applicability: DCPP Unit 1 operates as a base-load unit.

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

  DCPP 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 considered. There have been no modifications to reactor internals components at DCPP Unit 1 since May 2007.

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

4.3.3 WCAP-17451, Reactor Internals Guide Tube Wear – Westinghouse Domestic Fleet Operational Projections

In February 2015, the PWROG submitted Westinghouse topical report WCAP-17451-P (Reference 28) to the NRC for information only. The report documents the results of a PWROG program of which the purpose was 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 wear standpoint, as well as to establish an initial inspection schedule based on the various guide tube designs for the utilities that participated in the program.

As a result, the industry recognizes the document as providing the acceptance criteria for inspections of CRGT assembly guide cards.

4.3.4 Ongoing Industry Programs

The U.S. nuclear industry, through both the EPRI/MRP and the PWROG, continues to sponsor activities related to RVI aging management. PG&E will maintain cognizance of industry activities related to PWR internals inspection and aging management and will address and implement industry guidance stemming from those activities, as appropriate under NEI 03-08 practices.

4.4 SUMMARY

It should be noted that the PG&E, MRP, and 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 and then determination of the proper inspection or mitigation program that provides reasonable assurance that the components will continue to perform their intended functions through the period of extended operation. The GALL-based approach was used at DCPP for the initial basis of the LRA that resulted in the NRC SER (Reference 1).

The approach used to develop the DCPP Unit 1 AMP is fully compliant with regulatory directives and approved documents. The additional evaluations and analyses completed by the MRP industry group have provided clarification to 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 PG&E position that use of the AMR produced by the LRA methodology, 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 DIABLO CANYON POWER PLANT UNIT 1 REACTOR INTERNALS AGING MANAGEMENT PROGRAM ATTRIBUTES

The DCPP 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 (cracking)
- Primary water stress corrosion cracking (cracking)
- Irradiation-assisted stress corrosion cracking (cracking)
- Wear (loss of material)
- Fatigue (cracking)
- Thermal aging embrittlement (reduction in fracture toughness)
- Irradiation embrittlement (reduction in fracture toughness)
- Void swelling and irradiation growth (distortion)
- Thermal and irradiation-enhanced stress relaxation or irradiation-enhanced creep (loss of preload or loss of mechanical closure integrity)

The attributes of the DCPP Unit 1 Reactor Internals AMP and compliance with NUREG-1801 (GALL Report) and Section XI.M16A, "PWR Vessel Internals" (Reference 6), as updated via LR-ISG-2011-04 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.

PG&E fully utilized the GALL process contained in NUREG-1801 (Reference 6) in performing the aging management review of the reactor internals in the license renewal process. However, PG&E made several commitments for DCPP Unit 1 (see Reference 1), as discussed in Section 1. This Program Plan for the Inspection of Reactor Vessel Internals is to conform to LR-ISG-2011-04 prior to the period of extended operation at DCPP Unit 1. Additionally, PG&E committed to submitting responses to the applicable aging management program plant-specific action items, conditions and limitations identified in the NRC Safety Evaluation, Revision 1, on MRP-227 to the NRC by December 2015.

This AMP is consistent with the GALL process and includes consideration of the augmented inspections identified in MRP-227-A. The requirements of the commitment are hereby fulfilled. Specific details of the DCPP Unit 1 Reactor Internals AMP are summarized in the following subsections.

5.1 GALL REVISION 2 PROGRAM ELEMENT 1: SCOPE OF PROGRAM

GALL Report AMP Element Descriptions

> *"The scope of the program includes all RVI components based on the plant's applicable nuclear steam supply system. The scope of the program applies the methodology and guidance in MRP-227-A, which provides an 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 Babcock & Wilcox (B&W), Combustion Engineering (CE), and Westinghouse. The scope of components considered for inspection in MRP-227-A include core support structures, 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). In addition, ASME Code, Section XI includes inspection requirements for PWR removable core support structures in Table IWB-2500-1, Examination Category B-N-3, which are in addition to any inspections that are implemented in accordance with MRP-227-A.

The scope of the program does not include consumable items, such as fuel assemblies, reactivity control assemblies, and nuclear instrumentation. 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 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.'" (Reference 6).

Diablo Canyon Power Plant Unit 1 Program Scope

The DCPP PWR Vessel Internals Program provides guidelines to adequately manage the aging effects of selected DCPP reactor vessels internals components, both non-bolted and bolted. The DCPP Unit 1 RVI consist of three basic assemblies: (1) the upper core support structure that is removed during each refueling operation to obtain access to the reactor core, (2) the lower core support structure that can be removed, if desired, following a complete core unload, and (3) the incore instrumentation support structures. Additional RVI details are provided in subsection 4.2.2 of the DCPP Updated Final Safety Analysis Report (UFSAR) (Reference 22).

The DCPP PWR Vessel Internals Program will be focused on managing age related degradation mechanisms by performing inspections intended to identify crack initiation and growth due to IASCC. No additional aging management is necessary for the reactor vessel internals components in the No Additional Measures group. In no case does the No Additional Measures determination supersede the ASME Section XI Inservice Inspection requirements for components in this group.

The DCPP Unit 1 RVI subcomponents that required aging management review are indicated in the previously submitted Table 2.3.1-1 of the DCPP LRA (References 13 and 34). These components were subjected to an aging management review, and the results of this review were presented in Table 3.1.2-1 of the DCPP LRA. Specific columns of LRA Table 3.1.2-1 are reproduced in Appendix B as Table B-1, which includes all of the subcomponents of the RVI that required aging management review along with the related NUREG-1801 item(s) and the relevant Table 3.1.1 item from the LRA.

DCPP LRA Table 3.1.2-1 provides the detailed results of the reactor internals aging management review. The table identifies the aging effects that require management for those components requiring review. A column in the tables lists the programs and activities at DCPP Unit 1 that are credited to address the aging effects for each component 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 (Reference 1).

The results of the industry research provided by MRP-227-A, summarized in the tables in 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 DCPP Unit 1 RVI AMP scope is based on previously established and approved GALL Report approaches through application of the WCAP-14577 methodologies to determine those components that require aging management. Likewise, the additional information provided in the industry guidelines document, MRP-227-A (results of Appendix C), is rooted in the GALL methodology and provides a basis for augmented inspections that were required to complete this DCPP Unit 1 RVI AMP by providing the inspection method, frequency of inspection, and examination acceptance criteria.

Conclusion

This element complies with the corresponding aging management attribute in NUREG-1801, Section XI.M16A and License Renewal Commitments 72 and 73 (Reference 34).

5.2 GALL REVISION 2 PROGRAM ELEMENT 2: PREVENTIVE ACTIONS

GALL Report AMP Element Descriptions

> *"MRP-227-A 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, as described in GALL AMP XI.M2, 'Water Chemistry'."* (Reference 6)

Diablo Canyon Power Plant Unit 1 Preventive Actions

The DCPP PWR Vessel Internals Program does not prevent degradation due to aging effects; rather, it provides measures for monitoring to detect degradation prior to loss of intended function. Preventative measures to mitigate aging effects such as loss of material and cracking in the primary water system are established and implemented in accordance with the DCPP Water Chemistry Program (Reference 15).

Primary Water Chemistry Plan

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, etc.) 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 Diablo Canyon Power Plant Primary Strategic Water Chemistry Plan (Reference 15) is based on the current, approved revisions of EPRI PWR Primary Water Chemistry Guidelines (Reference 9).

This program is consistent with the corresponding program described in the GALL Report (References 6 and 22).

The limits of known detrimental contaminants imposed by the chemistry monitoring program are consistent with the EPRI PWR Primary Water Chemistry Guidelines (Reference 9).

Conclusion

This element complies with the corresponding aging management attribute in NUREG-1801, Section XI.M16A and License Renewal Commitments 72 and 73 (Reference 34).

5.3 GALL REVISION 2 PROGRAM ELEMENT 3: PARAMETERS MONITORED OR INSPECTED

GALL Report AMP Element Descriptions

> *"The program manages the following age-related degradation effects and mechanisms that are applicable in general to RVI components at the facility: (a) cracking induced by SCC, PWSCC, IASCC, or fatigue/cyclic 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 dimensions due to void swelling, or distortion; and (e) loss of preload due to thermal and irradiation-enhanced stress relaxation or creep.*
>
> *For the management of cracking, the program monitors the evidence of surface breaking linear discontinuities if a visual inspection technique is used as the non-destructive examination (NDE) method, or for relevant flaw presentation signals if a volumetric ultrasonic testing (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. Instead, the impact of loss of fracture toughness on component integrity is indirectly managed by: (1) using visual or volumetric examination techniques to monitor for cracking in the components and (2) applying applicable reduced fracture toughness properties in the flaw evaluations, in cases where cracking is detected in the components and is extensive enough to necessitate a supplemental flaw growth or flaw tolerance evaluation. The program uses physical measurements to monitor for any dimensional changes due to void swelling or distortion.*
>
> *Specifically, the program implements the parameters monitored/inspected criteria consistent with the applicable tables in Section 4, 'Aging Management Requirements,' in MRP-227-A."*(Reference 6)

Diablo Canyon Power Plant Unit 1 Parameters Monitored or Inspected

The DCPP PWR Vessel Internals program monitors the following aging effects by inspection, in accordance with the guidance of MRP-227-A or ASME Code Section XI, Category B-N-3:

1) Cracking

Cracking is due to SCC, PWSCC, IASCC, or fatigue/cyclical loading. Cracking is monitored with a visual inspection for evidence of surface-breaking linear discontinuities or a volumetric examination. Surface examinations may also be used to supplement visual examinations for detection and sizing of surface-breaking discontinuities.

2) Loss of Material

Loss of material is due to wear. Loss of material is monitored with a visual inspection for gross or abnormal surface conditions.

3) Loss of Fracture Toughness

Loss of fracture toughness is due to TE or IE. The impact of loss of fracture toughness on component integrity is indirectly managed by monitoring for cracking by using visual or volumetric examination techniques, and by applying applicable reduced fracture toughness properties in flaw evaluations if any detected cracking is determined to be extensive enough to necessitate a supplemental flaw growth or flaw tolerance evaluation.

4) Changes in Dimension

Changes in dimension are due to void swelling or distortion. The program supplements visual inspection with physical measurements to monitor for any dimensional changes due to void swelling or distortion.

5) Loss of Preload

Loss of preload is due to thermal and ISR or irradiation-enhanced creep. Loss of preload is monitored with a visual inspection for gross surface conditions that may be indicative of loosening in applicable bolted, fastened, keyed or pinned connections.

The DCPP PWR Vessel Internals program manages the aging effects noted above consistent with the guidance designated for the Westinghouse-designed Primary components included in Table 4-3 of MRP-227-A and the Westinghouse-designed Expansion components included in Table 4-6 of MRP-227-A.

For license renewal, the ASME Section XI Program consists of periodic volumetric, surface, and/or visual examination of components for assessment, signs of degradation, and corrective actions. This program is consistent with the corresponding program described in the GALL Report (Reference 6).

Appendices B and C of the DCPP Unit 1 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 and License Renewal Commitments 72 and 73 (Reference 34).

5.4 GALL REVISION 2 PROGRAM ELEMENT 4: DETECTION OF AGING EFFECTS

GALL Report AMP Element Descriptions

"The inspection methods are defined and established in Section 4 of MRP-227-A. Standards for implementing the inspection methods are defined and established in MRP-228. In all cases, well-established inspection methods are selected. These methods include volumetric UT examination methods for detecting flaws in bolting 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). VT-3 visual methods may be applied for the detection of cracking in non-redundant RVI components only when the flaw tolerance of the component, as evaluated for reduced fracture toughness properties, is known and the component has been shown to be tolerant of easily detected large flaws, even under reduced fracture toughness conditions. VT-3 visual methods are acceptable for the detection of cracking in redundant RVI components (e.g., redundant bolts or pins used to secure a fastened RVI assembly).

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.

The program adopts the guidance in MRP-227-A for defining the 'Expansion Criteria' that needed to be applied to the inspection findings of 'Primary' components and for expanding the examinations to include additional 'Expansion' components. RVI component inspections are performed consistent with the inspection frequency and sampling bases for 'Primary' components, 'Existing Programs' components, and 'Expansion' components in MRP-227-A.

In some cases (as defined in MRP-227-A), 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 dimensions due to void swelling or distortion.

Inspection coverages for 'Primary' and 'Expansion' RVI components are implemented consistent with Sections 3.3.1 and 3.3.2 of the NRC SE, Revision 1, on MRP-227." (Reference 6)

Diablo Canyon Power Plant Unit 1 Detection of Aging Effects

The DCPP PWR Vessel Internals program detects the aging effects listed in Element 3 through performance of examinations of the parameters specified in MRP-227-A, Table 4-3 for Westinghouse-designed Primary components and for parameters specified in MRP-227-A, Table 4-6 for Westinghouse-designed expansion components.

The DCPP PWR Vessel Internals program provides both examination acceptance criteria for conditions detected during inspection of Westinghouse-designed Primary components, as well as criteria that are applied to determine if scope expansion of examinations is required. When the examination acceptance criteria for the Westinghouse designed Primary components included in MRP-227-A, Table 4-3 are not met, the program requires expanding the scope of examinations to include the additional Westinghouse-designed Expansion components included in MRP-227-A, Table 4-6.

MRP-227-A included a fourth group of components designated as requiring No Additional Measures. The aging of these components was determined to be negligible relative to other reactor internals, and therefore, the program does not include any measures to monitor the effects of aging degradation in these components.

The inspections of the DCPP PWR Vessel Internals program are conducted as recommended in MRP-227-A. The program standards for examination methods, procedure content, and personnel qualifications are consistent with the requirements specified in MRP-228. Volumetric (UT) and visual (VT-3, EVT-1) examinations are used for detecting aging effects including general conditions, surface breaking discontinuities; and cracking caused by SCC, IASCC and fatigue.

VT-3 examinations are applied to detection of cracking only when the flaw tolerance of the component or affected assembly, as evaluated for reduced fracture toughness, has been shown to be tolerant of easily detectable large flaws, even under reduced fracture toughness conditions. VT-3 examinations may also be used to inspect for loss of material that is induced by wear, and other aging effects such as gross distortion caused by void swelling and irradiation growth, and aging effects of loss of preload that is caused by thermal and irradiation-enhanced stress relaxation and creep.

Surface measurements may be used to supplement visual examinations required by this program to reject or accept relevant indications.

The impact of loss of fracture toughness (due to TE or IE) on component integrity is indirectly managed by monitoring for cracking using visual or volumetric examination techniques, and by applying applicable reduced fracture toughness properties in the flaw evaluations after cracking is determined to be extensive enough to warrant a supplemental flaw growth or flaw tolerance evaluation.

One hundred percent of the accessible volume/area of each component will be examined for the Primary and Expansion components inspection category components. The minimum examination coverage for primary and expansion inspection categories is 75 percent of the component's total (accessible plus inaccessible) inspection area/volume be examined. When addressing a set of like components (e.g., bolting), the minimum examination coverage for primary and expansion inspection categories is 75 percent of the component's total population of like components (accessible plus inaccessible).

If conditions are detected during the examination, DCPP will enter the information into the corrective action program and evaluate whether the results of the examination ensure that the component (or set of components) will continue to meet the intended function under all licensing basis conditions of operation until the next scheduled examination. Engineering evaluations that demonstrate the acceptability of a detected condition will be performed consistent with WCAP-17096-NP.

Detection of indications required by the ASME Section XI ISI Program is well-established and field-proven through 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.

Inspection can be used to detect physical effects of degradation in both CASS and non-CASS components, 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 visual techniques include VT-3, and EVT-1 (enhanced visual test). 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 in augmented reactor internals inspections are documented in MRP-228.

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

In the augmented inspections detailed in the 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. This includes both CASS and non-CASS components. 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, "Reactor Internals Acceptance Criteria Methodology and Data Requirements" (Reference 12). 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 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 five 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 five 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.

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 the strategy proposed by MRP-227-A, UT inspections have been recommended exclusively for detection of flaws in bolts.

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 minimum acceptable bolting patterns. The evaluation program must demonstrate that the remaining bolts meet the requirements for a minimum 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 minimum acceptable bolting pattern before the next inspection.

Establishment of the minimum 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 inspection of bolts assumes that the plant owner will work with the designer to establish minimum acceptable bolting patterns prior to the inspection to support continued operation. For Westinghouse-designed plants, minimum acceptable bolting patterns for baffle-former and barrel-former bolts are available through the PWROG. PG&E has been a full participant in the development of the PWROG documents and has full access and use.

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 DCPP 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 and License Renewal Commitments 72 and 73 (Reference 34).

5.5 GALL REVISION 2 PROGRAM ELEMENT 5: MONITORING AND TRENDING

GALL Report AMP Element Descriptions

> *"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-A and its subsections. Flaw evaluation methods including recommendations for flaw depth sizing and for crack growth determinations as well as for performing applicable limit load, linear elastic and elastic-plastic fracture analyses of relevant flaw indications, are defined in MRP-227-A. The examination and re-examinations that are implemented in accordance with MRP-227-A guidance, together with the criteria specified in MRP-228 for inspection methodologies, inspection procedures, and inspection personnel, provide for timely detection, reporting, and implementation of corrective actions for the aging effects and mechanisms managed by the program.*
>
> *The program applies applicable fracture toughness properties, including reductions for thermal aging or neutron embrittlement, in the flaw evaluations of the components in cases where cracking is detected in a RVI component and is extensive enough to warrant a supplemental flaw growth or flaw tolerance evaluation.*
>
> *For singly-represented components, the program includes criteria to evaluate the aging effects in the inaccessible portions of the components and the resulting impact on the intended function(s) of the components. For redundant components (such as redundant bolts, screws, pins, keys, or fasteners, some of which are accessible to inspection and some of which are not accessible to inspection), the program includes criteria to evaluate the aging effects in the population of components that are inaccessible to the applicable inspection technique and the resulting impact on the intended function(s) of the assembly containing the components."* (Reference 6)

Diablo Canyon Power Plant Unit 1 Monitoring and Trending

The methods for monitoring, recording, evaluating and trending the date that result from the DCPP PWR Vessel Internals program's inspections are in accordance with the evaluation methodologies detailed in MRP-227-A, Section 6. This includes the recommended evaluation methodologies for flaw depth sizing and crack growth determinations, as well as for performing applicable limit load, linear elastic and elastic-plastic fracture analyses of relevant flaw indications.

The DCPP PWR Vessel Internals program applies applicable fracture toughness properties, including reductions for thermal aging or neutron embrittlement, in the flaw evaluations of the components in cases where cracking is detected in a reactor vessel internal component and is extensive enough to warrant a supplemental flaw growth or flaw tolerance evaluation.

In accordance with MRP-227-A, the DCPP PWR Vessel Internals program includes criteria to evaluate the aging effects in the inaccessible portions of the components and the resulting impact on the intended function(s) of the components. For redundant components, the program includes criteria to evaluate the aging effects in the population of components that are inaccessible to the applicable inspection technique and the resulting impact on the intended function(s) of the assembly containing the components.

Examination and re-examinations are implemented in accordance with MRP-227-A, together with the criteria 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 age-related degradation mechanisms within the scope of the program.

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 operating experience 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 are used to identify trends in materials degradation and forecast potential component degradation. The industry continues to share both material test data and operating experience through the auspices of the MRP and PWROG. PG&E has in the past and will continue to maintain cognizance of industry activities and will continue to share operating experience information related to PWR internals inspection and aging management.

Inspections credited in Appendix B are based on utilizing both the DCPP Unit 1 10-year ISI program and the augmented inspections derived from the industry program documented in MRP-227-A (Reference 3) and contained in Appendix C for reference purposes. 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.

Reporting requirements are included as part of the MRP-227-A guidelines (see subsection 4.3.2.2 of this AMP). 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 and License Renewal Commitments 72 and 73 (Reference 34).

5.6 GALL REVISION 2 PROGRAM ELEMENT 6: ACCEPTANCE CRITERIA

GALL Report AMP Element Descriptions

> *"Section 5 of MRP-227-A, which includes Table 5-1 for B&W-designed RVIs, Table 5-2 for CE-designed RVIs, and Table 5-3 for Westinghouse-designed RVIs, provides the specific examination and flaw evaluation acceptance criteria for the 'Primary' and 'Expansion' RVI component examination methods. For RVI components addressed by examinations performed in accordance with the ASME Code, Section XI, the acceptance criteria in IWB-3500 are applicable. For RVI components covered by other 'Existing Programs,' the acceptance criteria are described within the applicable reference document.*
>
> *As applicable, the program establishes acceptance criteria for any physical measurement monitoring methods that are credited for aging management of particular RVI components."* (Reference 6)

Diablo Canyon Power Plant Unit 1 Acceptance Criteria

The DCPP PWR Vessel Internals program acceptance criteria for the Westinghouse-designed Primary and Expansion component examinations are consistent with MRP-227-A, Section 5A. For the Westinghouse-designed Expansion components, ASME Code, Section XI, Section IWB-3500 acceptance criteria apply. The DCPP PWR Vessel Internals program establishes acceptance criteria for any physical measurement monitoring methods that are credited for aging management of particular reactor vessel internals components.

Those recordable indications that are the result of inspections required by the existing DCPP Unit 1 ISI program scope are evaluated in accordance with the applicable requirements of the ASME Code through the existing Corrective Action Program (Reference 23).

Inspection acceptance and expansion criteria are provided in 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 DCPP Unit 1 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, 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 a minimum bolting approach 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 (Reference 23), 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. Additional analysis to establish Appendix C expansion component evaluation criteria is being performed through the efforts of the PWROG. Status is monitored through direct PG&E 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 and XI.M16A and License Renewal Commitments 72 and 73 (Reference 34).

5.7 GALL REVISION 2 PROGRAM ELEMENT 7: CORRECTIVE ACTIONS

GALL Report AMP Element Descriptions

> *"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. The implementation of the guidance in MRP-227-A, 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 actions bases may be used to disposition relevant conditions if they have been previously approved or endorsed by the NRC. Alternative corrective actions not approved or endorsed by the NRC will be submitted for NRC approval prior to their implementation."* (Reference 6)

Diablo Canyon Power Plant Unit 1 Corrective Action

Any detected conditions that do not satisfy the examination acceptance criteria are required to be dispositioned through the DCPP corrective action program. The disposition will ensure that licensing and design basis functions of the reactor internals will be continued to be fulfilled.

The following corrective actions are suggested for the disposition of detected conditions that exceed the examination acceptance criteria:

1) Supplemental examinations to further characterize and potentially dispose of a detected condition

2) Engineering evaluation that demonstrates the acceptability of a detected condition

3) Repair, in order to restore a component with a detected condition to acceptance status

4) Replacement of a component with an unacceptable detected condition

If an engineering evaluation is used to disposition an examination result that does not meet the examination acceptance criteria, the engineering evaluation shall be conducted per an NRC-approved evaluation methodology.

DCPP QA procedures and review and approval processes are implemented in accordance with the requirements of 10 CFR 50, Appendix B and include administrative controls, as described in DCPP FSAR, Section 17.2, and provisions that specify when follow-up actions are required to be taken to verify that corrective actions are effective and those implemented to address significant conditions adverse to quality are effective in preventing recurrence of the condition.

The existing DCPP Unit 1 procedures for corrective actions (References 23, 24, and 25) and Inservice Repair and Replacement (Reference 26) will be credited for this element. The procedure in Reference 26 establishes the DCPP Unit 1 repair and replacement requirements for ASME Code Section XI, "Rules for Inservice Inspection of Nuclear Power Plant Components" (Reference 5). These requirements include the identification of a repair cycle, repair plan, and verification of acceptability for replacements. The corrective actions for augmented inspections at DCPP Unit 1 will be developed using repair and replacement procedures equivalent to those requirements in ASME B&PV Code, Section XI.

Conclusion

This element complies with the corresponding aging management attribute in NUREG-1801, Section XI.M16A and License Renewal Commitments 72 and 73 (Reference 34).

5.8 GALL REVISION 2 PROGRAM ELEMENT 8: CONFIRMATION PROCESS

GALL Report AMP Element Descriptions

> *"Site quality assurance procedures, review and approval processes, and administrative controls are implemented in accordance with the recommendations of NEI 03-08 and the requirements of 10 CFR Part 50, Appendix B, or their equivalent, as applicable. The implementation of the guidance in MRP-227-A, in conjunction with NEI 03-08 and other guidance documents, reports, or methodologies referenced in this AMP, provides and acceptable level of quality and an acceptance basis for confirming the quality of inspections, flaw evaluations, and corrective actions."* (Reference 6)

Diablo Canyon Power Plant Unit 1 Confirmation Process

DCPP QA procedures and review and approval processes are implemented in accordance with the requirements of 10 CFR 50, Appendix B and include administrative controls, as described in DCPP FSAR, Section 17.2, and provisions that specify when follow-up actions are required to be taken to verify that corrective actions are effective and those implemented to address significant conditions adverse to quality are effective in preventing recurrence of the condition.

The implementation of the guidance in MRP-227-A, in conjunction with the requirements of NEI 03-08 and other guidance documents, reports or methodologies referenced in this AMP, provides an acceptable level of quality and an acceptable basis for confirming the quality of inspection, flaw evaluation and other elements of aging management of the DCPP PWR Vessel Internals.

DCPP Unit 1 has an established 10 CFR Part 50, Appendix B, Program (Reference 27) that addresses the elements of corrective actions, confirmation process, and administrative controls. The DCPP 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. Recommendations from NEI 03-08 are considered in developing procedures and processes.

Conclusion

This element complies with or exceeds the corresponding aging management attribute in NUREG-1801, Section XI.M16A and License Renewal Commitments 72 and 73 (Reference 34).

5.9 GALL REVISION 2 PROGRAM ELEMENT 9: ADMINISTRATIVE CONTROLS

GALL Report AMP Element Descriptions

"The administrative controls for these types of programs, including their implementing procedures and review and approval processes, are implemented in accordance with the recommended industry guidelines and criteria in NEI 03-08, and are under existing site 10 CFR 50 Appendix B, Quality Assurance Programs, or their equivalent, as applicable. The evaluation in Section 3.5 of the NRC's SE, Revision 1, on MRP-227-A provides the basis for endorsing NEI 03-08. This includes endorsement of the criteria in NEI 03-08 for notifying the NRC of any deviation from the I&E methodology in MRP-227-A and justifying the deviation no later than 45 days after its approval by a licensee executive." (Reference 6)

Diablo Canyon Power Plant Unit 1 Administrative Controls

See the evaluation in Section 5.8.

Conclusion

This element complies with or exceeds the corresponding aging management attribute in NUREG-1801, Section XI.M16A and License Renewal Commitments 72 and 73 (Reference 34).

5.10 GALL REVISION 2 PROGRAM ELEMENT 10: OPERATING EXPERIENCE

GALL Report AMP Element Descriptions

> *"The review and assessment of relevant operating experience for impacts on the program, including implementing procedures, are governed by NEI 03-08 and Appendix A of MRP-227-A. Consistent with MRP-227-A, the reporting of inspection results and operating experience treated as a "Needed" category item under the implementation of NEI 03-08.*
>
> *The program is informed and enhanced when necessary through the systematic and ongoing review of both plant-specific and industry operating experience, as discussed in Appendix B of the GALL report, which is documented in LR-ISG-2011-05."* (Reference 6)

Diablo Canyon Power Plant Unit 1 Operating Experience

Extensive industry and DCPP Unit 1 operating experience 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" (Reference 29) and 98-11, "Cracking of Reactor Vessel Internal Baffle Former Bolts in Foreign Plants" (Reference 30). 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 operating experience 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 are expected to discover general internals structure degradation. To date, very little degradation has been observed industry wide.

The systematic and ongoing review and assessment of relevant DCPP-specific and industry operating experience for its impact to the program are governed by NEI 03-08, "Guideline for the Management of Materials Issues" and MRP-227-A, Appendix A.

Based on industry operating experience, DCPP Unit 1 proactively replaced the originally installed Alloy X-750 guide tube support pins (split pins) with strain hardened (cold worked) 316 stainless steel pins in 1999 to reduce the susceptibility for SCC in the support pins..

Relatively few incidents of PWR internals aging degradation have been reported in operating U.S. commercial PWR plants, and a summary of observations is maintained in Appendix A of MRP-227-A. With expectation of the ASME Section Inservice Inspection portions, the DCPP PWR Vessel Internals program is a new program. A key element of the program defined in MRP-227-A is the requirement for utilities to continue to report aging effects of PWR vessel internal components identified during examination. DCPP, through its participation in PWR Owners Group and EPRI-MRP activities, will continue to benefit from the reporting of examination information and results, and will share its own operating experience with the industry through those groups.

Industry operating experience is routinely reviewed by PG&E engineers using Institute of Nuclear Power Operations (INPO) Operating Experience (OE), the Nuclear Network, and other information sources, as directed under the applicable procedure (Reference 31 and 32), 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. PG&E will continue to 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.

Conclusion

This element complies with or exceeds the corresponding aging management attribute in NUREG-1801, Section XI.M16A and License Renewal Commitments 72 and 73 (Reference 34).

6 DEMONSTRATION

PG&E has demonstrated a long-term commitment to aging management of reactor internals at DCPP Unit 1. 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 DCPP Unit 1 reactor vessel internals have been performed during each 10-year interval since plant operations commenced.
- As documented in PG&E administrative procedures, reports are continuously reviewed by DCPP Unit 1 personnel with PG&E personnel for applicable issues that indicate operating procedures or programs require updates based on new OE.
- Review of Quality Verification Department audit reports, NRC inspection reports, and INPO evaluations indicate no unacceptable issues related to reactor vessel internals inspections.
- The Primary Water Chemistry Program (Reference 15) at DCPP Unit 1 has been effective in maintaining oxygen, halogens, and sulfate at levels sufficiently low to prevent SCC of the reactor vessel internals.
- The control rod guide tube support pins at DCPP Unit 1 were replaced (Reference 20).
- DCPP Unit 1 performed a power uprate (Reference 21) to increase its operating power to 3411 MWt.
- PG&E has actively participated in past and ongoing EPRI and PWROG RVI activities. PG&E will continue to 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.

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 degradation at its first appearance, consistent with the ASME 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.

DCPP Unit 1 will enter the period of extended operation at midnight on November 2, 2024 in conjunction with Cycle 24. In compliance with MRP-227-A requirements, the augmented inspections discussed in this AMP have been incorporated into the DCPP RVI AMP program procedure (Reference 43), along with other applicable DCPP procedures used to perform the ASME Section XI examinations, and implemented according to the requirements of MRP-227-A and Appendix C of this AMP. As discussed, the industry MRP-227-A guidelines also provide updates as experience is gained through inspection results. This feedback loop will enable updates based on actual inspection experience.

The DCPP Unit 1 Inspection Program Plan for Reactor Vessel Internals is comprised of the augmented inspections described in this document, as summarized in Appendix C, combined with the ASME Section XI ISI program inspections. In addition to existing DCPP Unit 1 programs and use of Operating Experience Reports (OERs), these inspections provide reasonable assurance that the reactor internals will continue to perform their intended functions through the period of extended operation.

6.1 DEMONSTRATION OF TOPICAL REPORT CONDITION COMPLIANCE TO SAFETY EVALUATION ON MRP-227, REVISION 0

Table 6-1 lists the compliance of topical report conditions to the safety evaluation (SE) on MRP-227.

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 (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 DCPP Unit 1.
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	In Table C-1 for the baffle-former bolts, the inspection frequency was changed from 10 to 15 additional effective full-power years (EFPY) to subsequent examination on a 10-year interval.
6. Periodicity of the re-examination of "Expansion" inspection category components	Applicable	"Re-inspection every 10 years following initial inspection" was added to every component under the Examination Method/Frequency column in Table C-2.
7. Updating of MRP-227, Revision 0, Appendix A	Applicable	Section 5 is updated to reflect XI.M16A from GALL Revision 2 (Reference 6).

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." (Reference 3)

DCPP Unit 1 Compliance

The process used to verify that DCPP 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:

1. Identify typical Westinghouse PWR internals components (MRP-191, Table 4-4).
2. Identify DCPP Unit 1 PWR internals components.
3. Compare typical Westinghouse PWR internals components to the DCPP Unit 1 PWR internals components.
    a. Confirm that no additional items were identified by this comparison (primarily supports A/LAI 2).
    b. Confirm that the materials identified for DCPP Unit 1 are consistent with those materials identified in MRP-191, Table 4-4.
    c. Confirm that the DCPP Unit 1 internals are the same as, or equivalent to, the typical Westinghouse PWR internals regarding design and fabrication.
4. Confirm that the DCPP Unit 1 operating history is consistent with the assumptions in MRP-227-A regarding core loading patterns.
5. Confirm that the DCPP Unit 1 RVI materials operated at temperatures within the original design basis parameters.
6. Determine stress values based on design basis documents.
7. Confirm that any changes to the DCPP Unit 1 RVI components do not impact the application of the MRP-227-A generic aging management strategy.

DCPP 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. DCPP Unit 1 operating history is consistent with the assumptions in MRP-227-A with regard to neutron fluence.
   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. The DCPP Unit 1 fuel management program changed from a high-leakage to a low-leakage core loading pattern prior to 30 years of operation. The first two cycles for DCPP Unit 1 were traditional core, with some feed assemblies on the periphery. Starting in cycle 3+, the core design changed to a low-leakage loading pattern. DCPP Unit 1 meets the fluence and fuel management assumptions in MRP-191 and the requirements for MRP-227-A application.
   b. DCPP Unit 1 has operated under base-load conditions over the life of the plant. Therefore, the actual number of unit loading and unloading occurrences is expected to be a small fraction of the cycles assumed in the fatigue analyses. Due to the infrequent nature of this cyclic transient, and due to the large margin to the assumed number of occurrences, it is not necessary to track the occurrence (Reference 35). Since DCPP Unit 1 operates at base load, assumptions in MRP documents regarding operational parameters affecting fluence are satisfied.
2. The DCPP Unit 1 RCS operates between T_{hot} and T_{cold}, which are not lower than 531.7°F for T_{cold} and not higher than 610.1°F for T_{hot} (Reference 22). The design temperature for the reactor vessel is 650°F (Reference 22). DCPP Unit 1 operating history is within original design basis parameters; therefore, it is consistent with the assumptions used to develop the MRP-227-A aging management strategy with regard to temperature operational parameters.
3. DCPP Unit 1 internals components and materials are comparable to the typical Westinghouse PWR internals components (MRP-191, Table 4-4).
   a. No additional components that adversely affect the MRP-191 FMECA process are identified for Diablo Canyon Unit 1 and the components required to be in the DCPP Unit 1 RVI program (Reference 13) are consistent with those contained in MRP-191 (Reference 10).
   b. Materials for DCPP Unit 1 are consistent with, or nearly equivalent to, those materials identified in MRP-191, Table 4-4 for Westinghouse-designed plants (Reference 10). The exceptions are: the guide plates/cards; the mixing devices; the upper instrumentation brackets, clamps, terminal blocks, and conduit straps (conduit support, conduit support gusset, gusset clamp, and thermocouple stop); and the BMI column cruciform material, which are identified as having a material different that that specified in MRP-191 and involve CF8. An expert panel was conducted to disposition these component material differences (Reference 36). Several additional components have different materials than those specified in MRP-191; however, they have been determined to have no effect of the recommended MRP aging management inspection sampling strategy.
   c. DCPP Unit 1 internals are the same as, or equivalent to, the typical Westinghouse PWR internals regarding design and fabrication.

4. Modifications to the DCPP Unit 1 reactor internals include a flux thimble tube inspection and replacement program (Reference 18), a control rod guide tube support pin replacement project (Reference 20), and a power uprating project (Reference 21). Details of these replacements are retained in plant records. No additional modifications were made to the plant after 2011. The design has been maintained over the lifetime of the plant as specified by the original equipment manufacturer, operational parameters are compliant with MRP-227-A requirements with regard to fluence and temperature, and the components and materials are the same as those considered in MRP-191. Therefore, the DCPP 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 DCPP Unit 1 evaluation for A/LAI 1 of the NRC SE on MRP-227, Revision 0, confirms that MRP-227-A is applicable to DCPP Unit 1. An expert panel was held that concluded that the material differences for components within the DCPP Unit 1 RVI program did not result in any additional measures or changes to the MRP-227-A aging management program for DCPP Unit 1.

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." (Reference 3)

DCPP Unit 1 Compliance

This A/LAI requires comparison of the RVI components that are within the scope of license renewal for Diablo Canyon Unit 1 to those components contained in MRP-191, Table 4-4. DCPP Unit 1 RVI components were tabulated (References 13 and 34) and compared to the typical Westinghouse PWR components in MRP-191, Table 4-4. From the review, it was determined that the components required to be in the DCPP Unit 1 program (Reference 13) are consistent with those contained in MRP-191.

Several components have different materials than those specified in the MRP-191 assessment. The potential for alternate materials, specifically CF8, to be used for the guide plates/cards and the brackets, clamps, terminal blocks, and conduit straps was identified. An expert panel was conducted to disposition these material differences (Reference 36). It was concluded that the material differences result in No Additional Measures or changes to the existing aging management program and MRP-227-A inspection schedule.

The completion of the expert panel supports the requirement that the AMP shall provide assurance that the effects of aging on the DCPP 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. Several additional components have slightly different materials specifications (i.e., different grades of austenitic stainless steel) than those specified in MRP-191; however, they have been determined to have no effect on the recommended MRP aging management inspection strategy.

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, are applicable to DCPP Unit 1 with no modifications.

Conclusion

DCPP Unit 1 complies with A/LAI 2 of the NRC SE on MRP-227, Revision 0, for all components as a result of the conclusion of the expert panel.

Therefore, DCPP Unit 1 meets the requirement for application of MRP-227-A as a strategy for managing age-related material 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." (Reference 3)

DCPP Unit 1 Compliance

DCPP Unit 1 is compliant with the requirements in MRP-227-A, Table 4-9, as shown in Table C-3 of this document. This is detailed in the plant-specific DCPP program documents for ASME Section XI (Reference 2) and the plant-specific flux thimble program (Reference 18).

Conclusion

DCPP 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." (Reference 3)

DCPP Unit 1 Compliance

This Applicant/Licensee Action Item is not applicable to DCPP Unit 1 since it 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 DCPP 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." (Reference 3)

DCPP Unit 1 Compliance

See Table 7-1. DCPP Unit 1 utilizes a Type 304 SS hold-down spring; therefore, PG&E is planning to perform inspections/physical measurements on the DCPP Unit 1 hold-down spring according to MRP-227-A.

An evaluation was performed by Westinghouse to develop acceptance criteria for the inspection/physical measurement of the hold-down spring that will ensure the hold-down spring remains capable of performing its required functions.

Conclusion

DCPP 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."* (Reference 3)

DCPP Unit 1 Compliance

This Applicant/Licensee Action Item is not applicable to DCPP Unit 1 since it 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 DCPP 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."* (Reference 3)

DCPP Unit 1 Compliance

The NRC final SE on MRP-227 (Reference 3, subsection 3.3.) states that for assessment of CASS materials, the applicant/licensee for license renewal may apply the criteria in the NRC letter of May 19, 2000, "License Renewal Issue No. 98-0030, Thermal Aging Embrittlement of Cast Austenitic Stainless Steel Components" (Reference 38) 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 components' 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 DCPP Unit 1 control rod guide tube assemblies have parts where alternate material ASTM International Standard A351, Grade CF8 was permitted in the fabrication. These parts are the intermediate flanges, lower flanges, and guide plates/cards on the lower guide tube assembly. The historical records showed that they were fabricated from wrought material; therefore, Applicant/Licensee Action Item 7 (A/LAI 7) is not applicable to Diablo Canyon Unit 1 control rod guide tubes.

The DCPP Unit 1 stand-alone mixing devices, upper internals instrumentation conduit and supports – stops, supports, gussets, and clamps; upper support column assemblies – bases (mixer and orifice base); lower support column bodies – caps; and lower support casting, are CASS materials.

For completeness, it is noted that the DCPP Unit 1 BMI column assemblies – cruciform were fabricated from wrought material.

For each of the CASS components, the elemental percentages from the chemical data retrieved from certified material test reports (CMTRs) for the CASS component are input into Hull's formula (per guidance of NUREG/CR-4513 (Reference 39) to calculate the delta ferrite content of the CASS material. The CMTRs do not list the element percentage for nitrogen; thus, per the guidance of NUREG/CR-4513, nitrogen is assumed to be 0.04 percent (Reference 39). The CMTRs do not typically list an elemental percentage for molybdenum. CASS materials A351 and A296, Grade CF8, did not have a requirement for percent molybdenum in 1968 (Reference 40). The 2013 Edition of the ASME Boiler & Pressure Vessel Code (Reference 41) has SA-351, Grade CF8 chemistry requirements that specify a maximum of 0.5 percent molybdenum; thus, this maximum value is conservatively input into Hull's formula. Where CMTRs were not located, a conservative combination of SA A351, Grade CF8 chemical requirements was used to show the ferrite content can potentially exceed 20 percent. The results of the TE evaluation for the DCPP Unit 1 CASS components are summarized in Table 6-2.

Based on the criteria of the NRC letter dated May 19, 2000 (Reference 38):

- The upper internals mixing devices are considered to be potentially susceptible to TE.
- The upper instrumentation conduit and supports (stops, supports, gussets, and clamps) are considered to be potentially susceptible to TE.
- For the upper support column – bases (mixing style), sixteen have ferrite content less than 20 percent, two have delta ferrite content that exceeds 20 percent, and one has ferrite content that potentially exceeds 20 percent. Thus, sixteen are not susceptible to TE and three are potentially susceptible to TE.
- For the upper support column – bases (orifice style), 26 are not susceptible to TE and three are potentially susceptible to TE.
- The lower support column bodies – caps are not susceptible to TE.
- The lower support casting is not susceptible to TE.

All of the preceding components were considered in MRP-191 (Reference 10) and were screened for susceptibility to material degradation, including consideration of TE and IE. With the exception of the upper instrumentation conduit and supports (stops, supports, gussets, and clamps), the aforementioned components were screened as CASS and were considered for TE in MRP-191. As discussed in the response to A/LAI 2, the assessments of the CASS upper instrumentation conduit and supports (stops, supports, gussets, and clamps) were evaluated by an expert panel, taking into consideration their potential susceptibility to TE and their impact on the Diablo Canyon Unit 1 aging management strategy. The expert panel concluded that the material differences result in No Additional Measures or changes to the existing aging management program and MRP-227-A inspection schedule.

No martensitic stainless steel or martensitic precipitation hardenable stainless steel components were identified for the DCPP Unit 1 reactor vessel internals.

Table 6-2 Summary of Diablo Canyon Unit 1 CASS Components and Their Susceptibility to TE

CASS Component MRP-191 [10] Name	Material	Molybdenum Content (Percent)	Casting Method	Ferrite Content (Percent)	Susceptibility to TE (Based on NRC Letter [38])
Upper Internals Assembly					
Mixing Devices	ASTM A351, Grade CF8	0.5 Maximum	Static(4)	Possible > 20%(2)	Potentially Susceptible(2)
Upper Instrumentation Conduit and Supports (Thermocouple Stops, Supports, Gussets and Clamps(5))	ASTM A240, A479, or ASTM A351, Grade CF8	0.5 Maximum	Static(4)	Possible > 20%(2)	Potentially Susceptible(2)
Upper Support Column Assemblies, Mixer Bases	ASTM A351, Grade CF8	0.5 Maximum	Static(4)	16 of 19 ≤ 20%(1)	16 of 19 Not Susceptible(1)
				2 of 19 >20% 1 of 19 Possible > 20%(2)(3)	3 of 19 Potentially Susceptible(2)
Upper Support Column Assemblies, Column Bases	ASTM A351, Grade CF8	0.5 Maximum	Static(4)	26 of 29 ≤ 20%(1)	Not Susceptible(1)
				3 of 29 Possible > 20%(2)(3)	Potentially Susceptible(2)
Lower Internals Assembly					
Lower Support Column Bodies – Caps	ASTM A296, Grade CF8	0.5 Maximum	Static(4)	≤ 20%(1)	Not Susceptible(1)
Lower Support Casting	ASTM A351, Grade CF8	0.5 Maximum	Static(4)	≤ 20%(1)	Not Susceptible(1)

Notes:

(1) Conclusion is based on CMTR chemistry data with molybdenum = 0.5 percent (based on ASTM International chemistry requirements) and nitrogen = 0.04 percent (per guidance of [39]) input into Hull's formula.

(2) When a CMTR was not located, a conservative combination of ASME SA-351, Grade CF8 chemical requirements input into Hull's formula [39] shows that ferrite content can potentially exceed 20 percent.

(3) A letter in the historical records provides certification that the original actual test result data were not available due to loss of specific traceability, because of the administrative systems used at the time of manufacturing. Thus, actual chemistry data are not available for input into Hull's formula and the cast part is conservatively assessed per note (2).

(4) The part is assumed to have been static cast.

(5) The Diablo Canyon Unit 1 CASS parts on the upper instrumentation conduit and supports are the thermocouple stops, supports, gussets, and clamps.

Conclusion

It is concluded that continued application of the MRP-227-A (Reference 3) strategy will meet the requirement for managing age-related degradation of the Diablo Canyon Unit 1 CASS reactor vessel internals components.

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." (Reference 3)

DCPP Unit 1 Compliance

MRP-227-A identifies the following information that license renewal applicants must submit to the NRC for review and approval:

1. *An AMP for the facility that addresses the 10 program elements as defined in NUREG-1801, Revision 2, AMP XI.M16A.*

   DCPP Unit 1: The 10 programs elements are addressed via issuance of this AMP.

2. *To ensure the MRP-227 program and the plant-specific action items will be carried out by applicants/licensees, applicant/licensees are to submit an inspection plan which addresses the identified plant-specific action items for staff review and approval consistent with the licensing basis for the plant. If an applicant/licensee plans to implement an AMP which deviates from the guidance provided in MRP-227, as approved by the NRC, the applicant/licensee shall identify where the program deviates from the recommendations in MRP-227, as approved by the NRC, and shall provide a justification for any deviation which includes a consideration of how the deviation affects both "Primary" and "Expansion" inspection category components.*

   DCPP Unit 1: The inspection plan for DCPP Unit 1 is being submitted via this AMP.

3. *The regulation at 10 CFR 54.21(d) requires that an FSAR supplement for the facility contain a summary description of the programs and activities for managing the effects of aging and the evaluation of TLAAs for the period of extended operation. Those applicants for LR referencing MRP-227, as approved by the NRC, for their RVI component AMP shall ensure that the programs and activities specified as necessary in MRP-227, as approved by the NRC, are summarily described in the FSAR supplement.*

   DCPP Unit 1: Per the LRA (References 1 and 34), PG&E will maintain a summary list in the DCPP FSAR Update of activities that are required to manage the effects of aging for the systems, structures and components in the scope of the license renewal during the period of extended operation. The FSAR update will comply with 10 CFR 54.21(d).

4. *The regulation at 10 CFR 54.22 requires each applicant for LR to submit any technical specifications changes (and the justification for the changes) that are necessary to manage the effects of aging during the period of extended operation as part of its LRA. For the plant CLBs that include mandated inspection or analysis requirements for RVI either in the operating license for the facility or in the facility TS, the applicant/licensee shall compare the mandated requirements with the recommendations in the NRC-approved version of MRP-227. If the mandated requirements differ from the recommended criteria in MRP-227, as approved by the NRC, the conditions in the applicable license conditions or TS requirements take precedence over the MRP recommendations and shall be complied with.*

   DCPP Unit 1: Per the LRA (References 1 and 34), no technical specifications changes have been identified in order to manage the effects of aging of RVI during the period of extended operation.

5. *Pursuant to 10 CFR 54.21(c)(1), the applicant is required to identify all analyses in the CLB for their RVI components that conform to the definition of a TLAA in 10 CFR 54.3 and shall identify these analyses as TLAAs for the application in accordance with the TLAA identification requirement in 10 CFR 54.21(c)(1). MRP-227 does not specifically address the resolution of TLAAs that may apply to applicant/licensee RVI components. Hence, applicants/licensees who implement MRP-227, as approved by the NRC, shall still evaluate the CLB for their facilities to determine if they have plant-specific TLAAs that shall be addressed. If so, the applicant's/licensee's TLAA shall be submitted for NRC review along with the applicant/licensee's application to implement the NRC-approved version of MRP-227.*

   For those cumulative usage factor (CUF) analyses that are TLAAs, the applicant may use the PWR Vessel Internals Program as the basis for accepting these CUF analyses in accordance with 10 CFR 54.21(c)(1)(iii) only if the RVI components in the CUF analyses are periodically inspected for fatigue-induced cracking in the components during the period of extended operation. The periodicity of the inspections of these components shall be justified to be adequate to resolve the TLAA. Otherwise, acceptance of these TLAAs shall be done in accordance with either 10 CFR 54.21(c)(1)(i) or (ii), or in accordance with 10 CFR 54.21(c)(1)(iii) using the applicant's program that corresponds to NUREG-1801, Revision 2, AMP X.M1, "Metal Fatigue of Reactor Coolant Pressure Boundary Program." To satisfy the evaluation requirements of ASME Code, Section III, Subsection NG-2160 and NG-3121, the existing fatigue CUF analyses shall include the effects of the reactor coolant system water environment.

   DCPP Unit 1: Per the LRA (References 1 and 34) there are no analyses in the CLB for RVI that are TLAAs, in accordance with 10 CFR 54.3(a) Criteria 2 and 3.

DCPP Unit 1, per the Regulatory Issue Summary (Reference 37), is considered a Category D plant that is expected to submit their RVI AMP based on the guidance of MRP-227-A, which is consistent with their commitments. Per License Renewal commitment (Reference 34), DCPP Unit 1 must submit responses to MRP-227-A actions items to the NRC by December 2015. DCPP Unit 1 must implement the PWR Reactor Internals Program to conform to LR-ISG-2011-04 prior to the period of extended operation.

7 PROGRAM ENHANCEMENT AND IMPLEMENTATION SCHEDULE

The requirements of MRP-227 are based on an 18-month refueling cycle and consider both effective full-power years (EFPY) and cumulative operation. The information contained in Table 7-1 is based on this information and includes a description of the currently projected scope of inspection pertaining to the reactor internals AMP. Should a change occur in plant operational practices or operating experience result in changes to the projections, appropriate updates will be performed on affected plant documentation in accordance with approved procedures. The results contained in Table 7-1 are based on average 20-month cycles in calculating the EFPY.

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

RFO	Cycle End Quarter/Year	Estimated EFPY	AMP-Related Scope	Inspection Method and Criteria	Comments
1R17	Spring / 2012	23.14			
1R18	Spring / 2014	24.73	ASME Code Section XI ISI	ASME Code Section XI	
1R19	Fall / 2015	26.26			
1R20	Spring / 2017	27.75	Initial MRP-227 augmented inspection for control rod guide tube guide cards[(2)]	WCAP-17451-P (Reference 28) is applied for control rod guide card inspections.	The initial inspection window for these components is no later than two refueling outages from the beginning of extended operation. DCPP Unit 1 has the option to perform these inspections until RO-26.
1R21	Spring / 2019	29.38			
1R22	Fall / 2020	30.93			
1R23	Spring / 2022	32.42			

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

RFO	Cycle End Quarter/Year	Estimated EFPY	AMP-Related Scope	Inspection Method and Criteria	Comments
1R24	Spring / 2024	34.06			Extended operation begins 11/2/2024
			ASME Code Section XI ISI	ASME Code Section XI	
			Initial MRP-227 augmented inspection for baffle-former bolts completed during or before this outage.	MRP-227 inspections in accordance with MRP-228 specifications.	Initial MRP-227 augmented inspection for baffle-former bolts must be completed between 25 and 35 EFPY.
			Initial MRP-227 augmented inspection for control rod guide tube lower flanges, upper and lower core barrel flange weld, upper and lower core barrel cylinder girth welds, and thermal shield flexures completed during or before this outage.	MRP-227 inspections in accordance with MRP-228 specifications.	The initial inspection window for these components is no later than two refueling outages from the beginning of extended operation. DCPP Unit 1 has the option to perform these inspections until RO-26.
			Initial MRP-227 augmented inspections for baffle-former assembly and baffle-edge bolts completed during or before this outage.	MRP-227 inspections in accordance with MRP-228 specifications.	The initial inspection window for the baffle-edge bolts and the baffle-former assembly is between 20 and 40 EFPY. DCPP Unit 1 has the option to perform these inspections until RO-27.
			Initial MRP-227 augmented inspection for internals hold down spring completed during or before this outage.	MRP-227 inspections in accordance with MRP-228 specifications.	The initial inspection window for the hold-down spring is within three cycles of the beginning of the license renewal period. DCPP Unit 1 has the option to perform this inspection until RO-27.
1R25	Fall / 2025	35.61			

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

RFO	Cycle End Quarter/Year	Estimated EFPY	AMP-Related Scope	Inspection Method and Criteria	Comments
1R26	Spring / 2027	37.10			
1R27	Spring / 2029	38.74			
1R28	Fall / 2030	40.29			
1R29	Spring / 2032	41.78			

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

RFO	Cycle End Quarter/Year	Estimated EFPY	AMP-Related Scope	Inspection Method and Criteria	Comments
1R30	Spring / 2034	43.42	ASME Code Section XI ISI	ASME Code Section XI	
			Subsequent MRP-227 augmented inspection for baffle-former bolts completed during or before this outage. (1)	MRP-227 inspections in accordance with MRP-228 specifications	The subsequent inspection window for these components is ten years after initial inspection.
			Subsequent MRP-227 augmented inspection for control rod guide tube guide cards, control rod guide tube lower flanges, upper core barrel flange weld, and thermal shield flexures completed during or before this outage. (1)	MRP-227 inspections in accordance with MRP-228 specifications.	The subsequent inspection window for these components is ten years after initial inspection.
			Subsequent MRP-227 augmented inspections for baffle-former assembly and baffle-edge bolts completed during or before this outage. (1)	MRP-227 inspections in accordance with MRP-228 specifications.	The subsequent inspection window for these components is ten years after initial inspection.
1R31	Fall / 2035	44.97			
1R32	Spring / 2037	46.46			
1R33	Spring / 2039	48.10			
1R34	Fall / 2040	49.64			
1R35	Spring / 2042	51.14			
1R36	Spring / 2044	52.78			Renewed Operating License expires 11/2/2044

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

RFO	Cycle End Quarter/Year	Estimated EFPY	AMP-Related Scope	Inspection Method and Criteria	Comments
Notes:					
1. Re-examination frequency is on a 10-year basis.					
2. Subsequent examination is dependent on results from the initial inspection (see Section 6 of WCAP-17451-P).					

8 IMPLEMENTING DOCUMENTS

The DCPP Unit 1 AMP also references the Water Chemistry Guidelines, Flux Thimble Tube Inspection program, and the ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD Program, which will become part of the DCPP Unit 1 Inspection Program Plan for Reactor Vessel Internals. MRP-227 augmented examinations (Appendix C), recommended as a result of industry programs, have been appropriately incorporated into the DCPP RVI AMP program procedure (Reference 43). The Program Plan for Inspection of Reactor Vessel Internals will coordinate with the existing DCPP Unit 1 Inservice Inspection Program and will supplement that program with the augmented examinations for managing the potential aging effects of the reactor vessel internals.

DCPP Unit 1 documents associated with the existing DCPP Unit 1 programs and considered to be implementing documents of the Reactor Vessel Internals Program are:

- Diablo Canyon Power Plant Water Chemistry Guidelines (References 15 and 16)
- Inservice Inspection Program Implementation (Reference 2)
- Flux Thimble Tube Inspection Program (References 17 and 19)

The Reactor Vessel Internals Program relies on the Water Chemistry Program for maintaining high water purity to reduce susceptibility to cracking due to SCC. The Water Chemistry Plan was evaluated and found to be consistent with GALL Section XI.M2, Water Chemistry (Reference 6). Additional procedures may be updated or created as OE for augmented examinations is accumulated.

Based on this information, the updated AMP for DCPP 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

1. U.S. Nuclear Regulatory Commission, ADAMS Accession No. ML 11153A103, "Safety Evaluation Report Related to the License Renewal of Diablo Canyon Nuclear Power Plant, Units 1 and 2," Docket Nos. 50-275 and 50-323. Pacific Gas and Electric Company, June 2011.

2. Diablo Canyon Power Plant Procedure AD5.ID2, "Inservice Inspection Program."

3. *Materials Reliability Program: Pressurized Water Reactor Internals Inspection and Evaluation Guidelines (MRP-227-A).* EPRI, Palo Alto, CA: 2011. 1022863.

4. U.S. Nuclear Regulatory Commission, 10 CFR Part 54, "Requirements for Renewal of Operating Licenses for Nuclear Power Plants," last updated July 6, 2012.

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21. Diablo Canyon Power Plant Design Change Package, DCP N-49516, Rev. 0, "Design Change Package."

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24. Diablo Canyon Power Plant Procedure OM7.ID1, "Problem Identification and Resolution."

25. Diablo Canyon Power Plant Procedure OM7.ID13, "Technical Evaluations."

26. Diablo Canyon Power Plant Procedure MA1.ID13, "ASME Section XI Repair/Replacement Program and Implementation."

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APPENDIX A
ILLUSTRATIONS

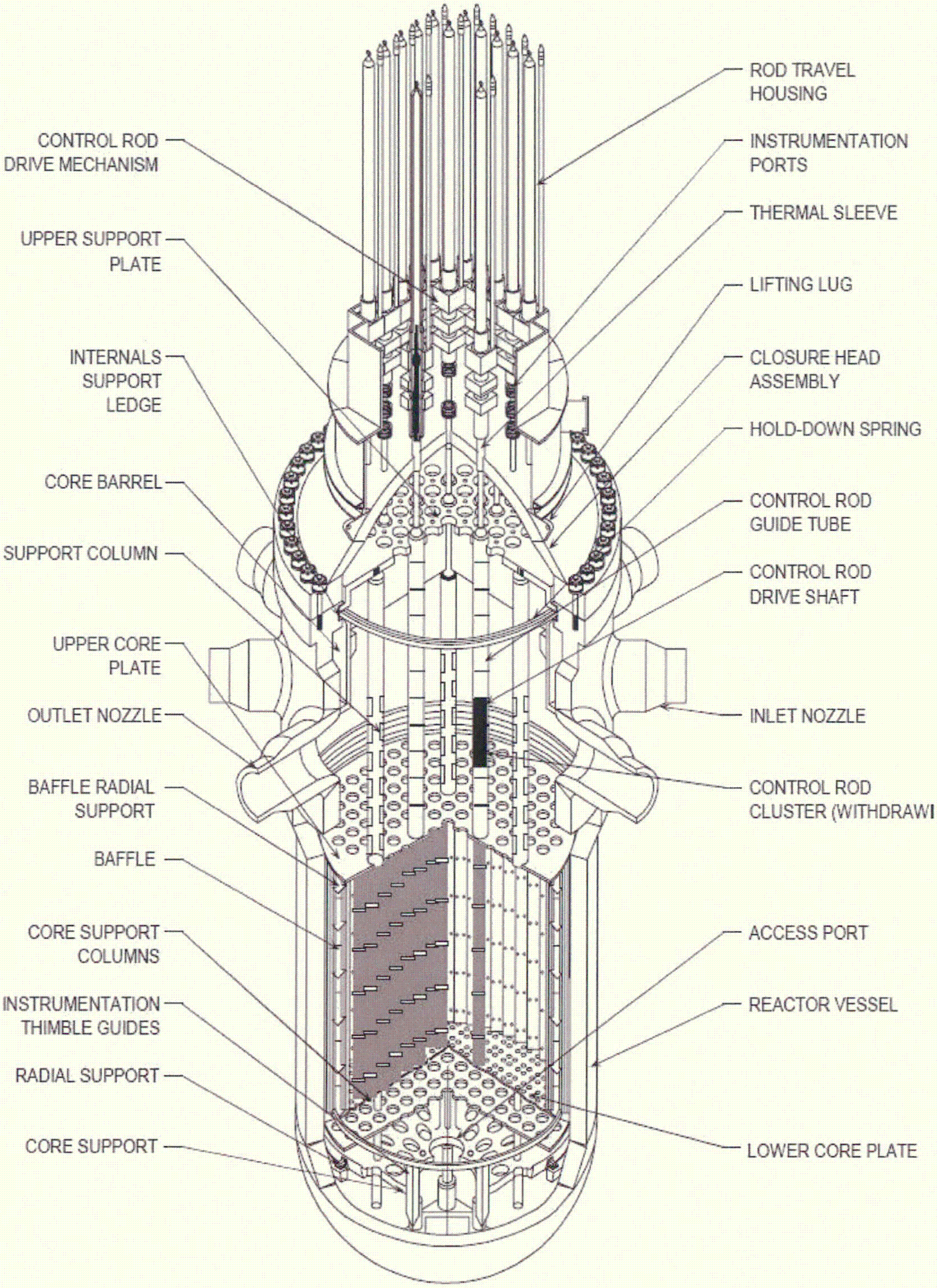

Figure A-1 Illustration of Typical Westinghouse Internals

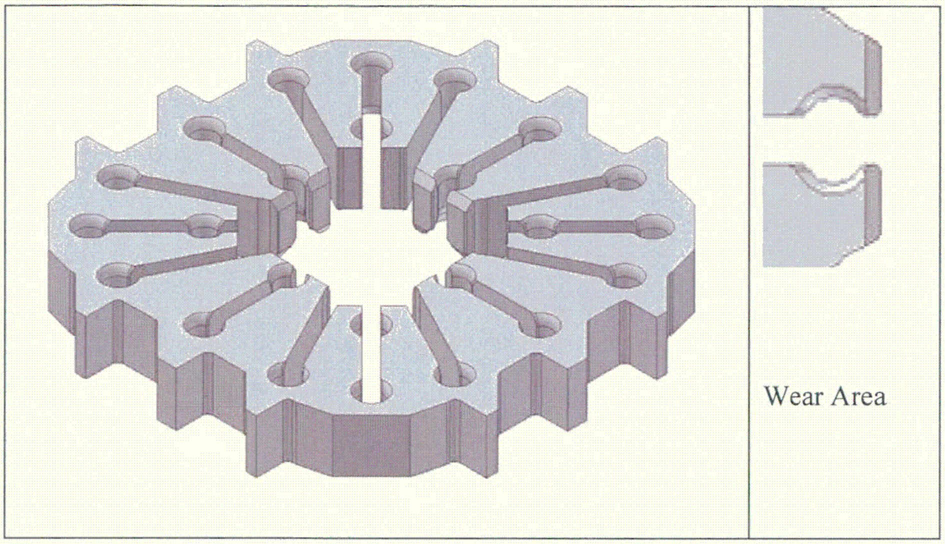

Figure A-2 Typical Westinghouse Control Rod Guide Card

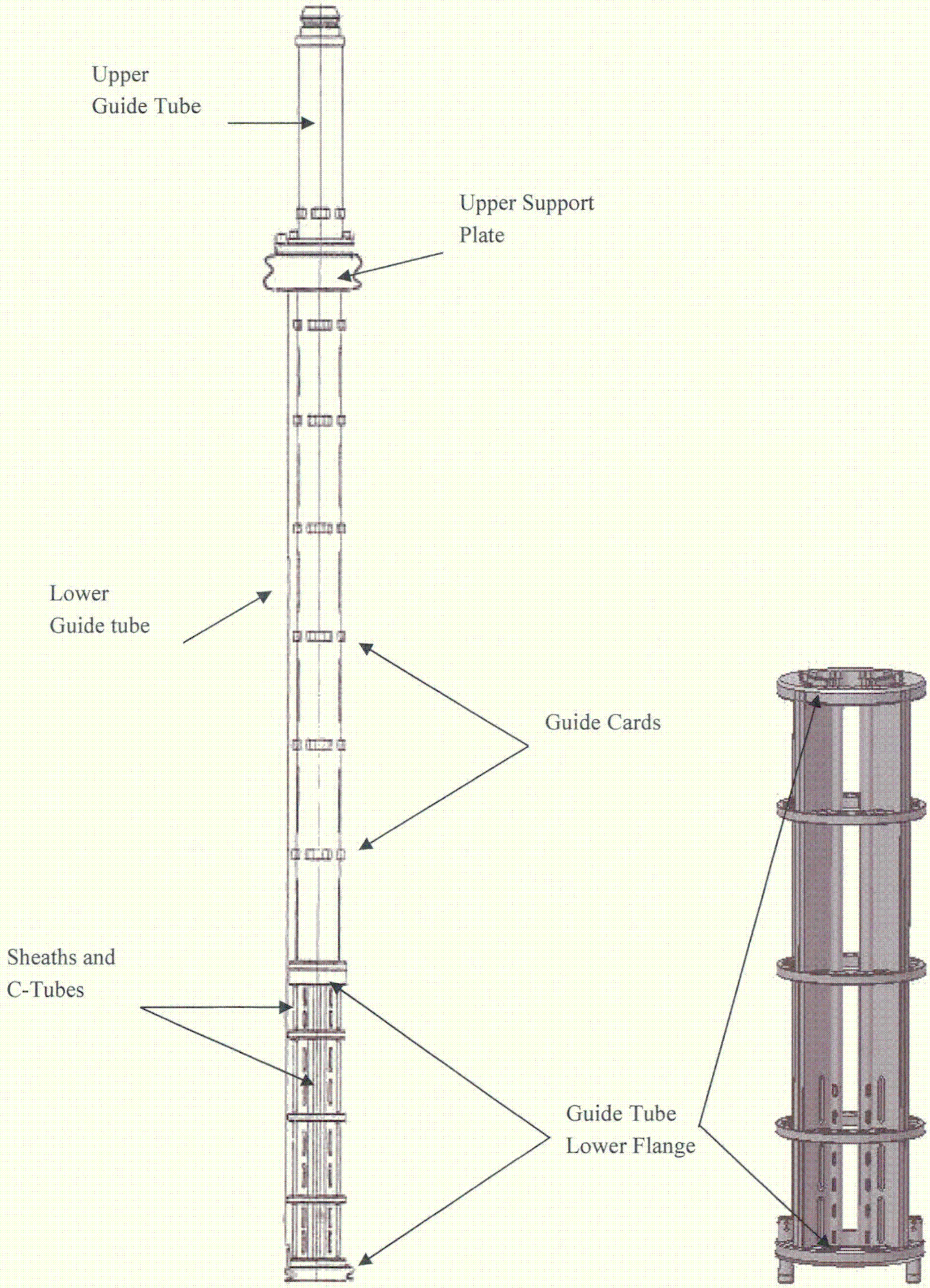

Figure A-3 Typical Lower Section of Control Rod Guide Tube Assembly

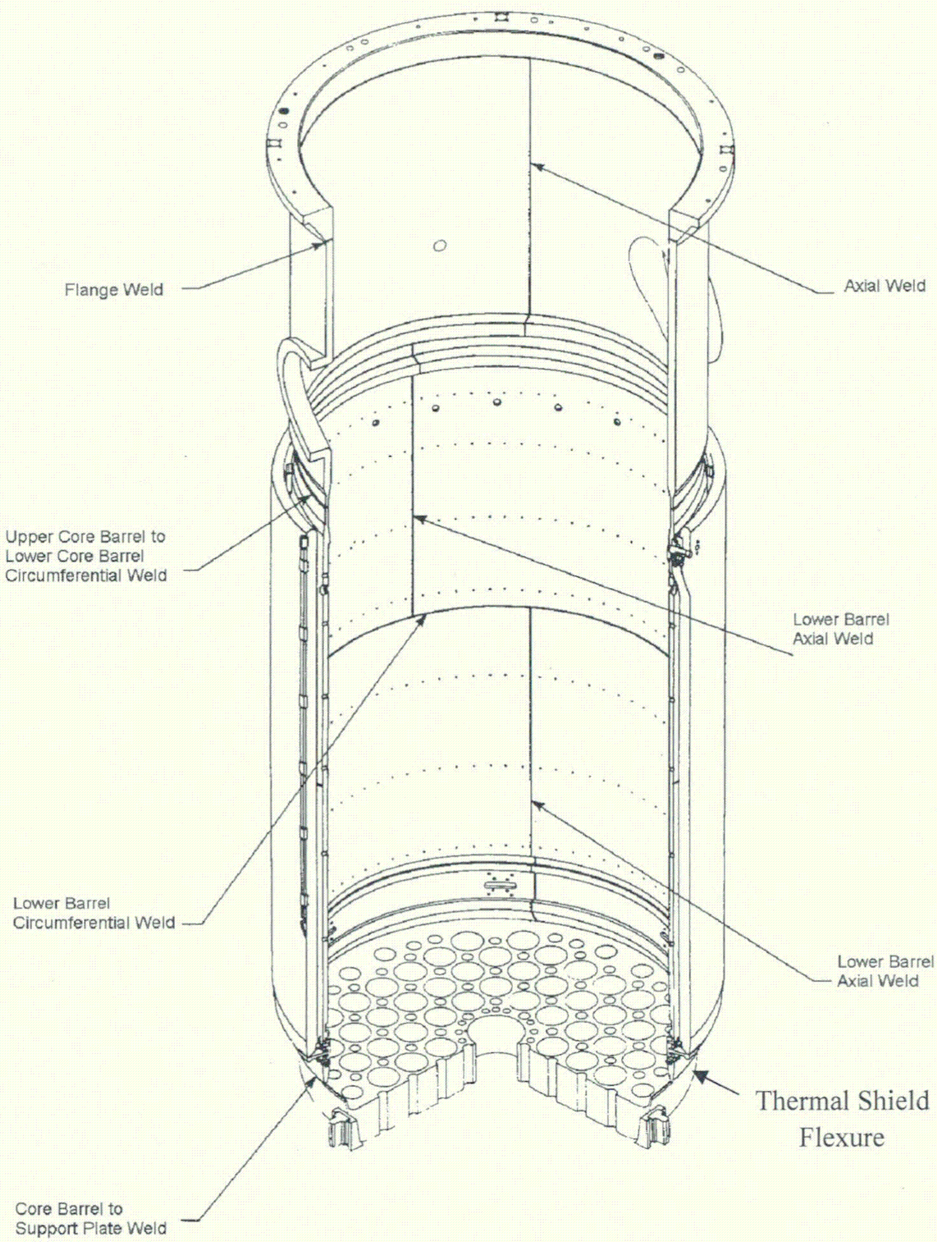

Figure A-4 Major Core Barrel Welds

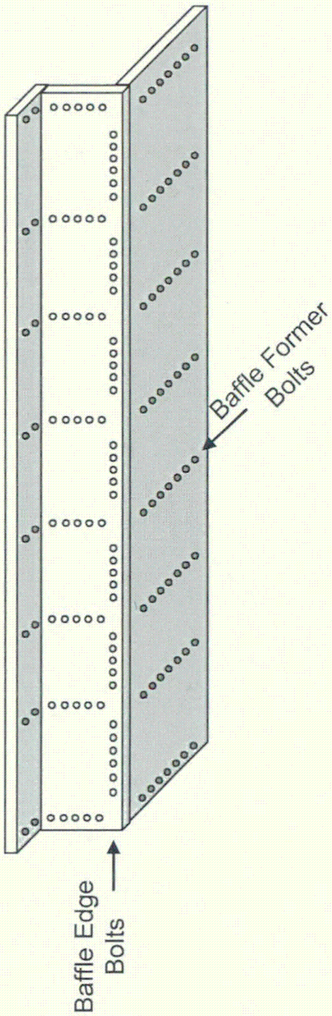

Figure A-5 Bolting Systems Used in Westinghouse Core Baffles

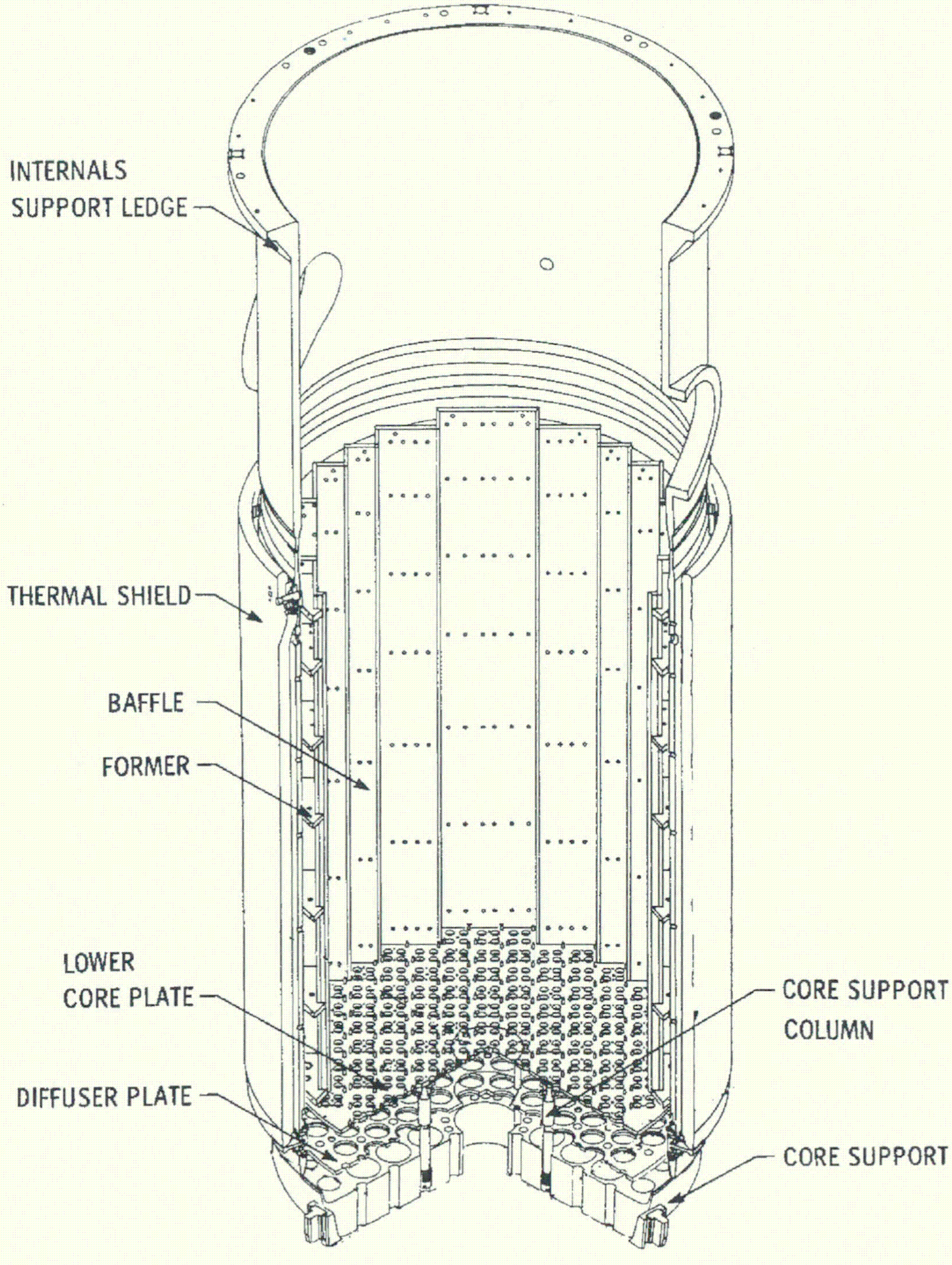

Figure A-6 Core Baffle/Barrel Structure

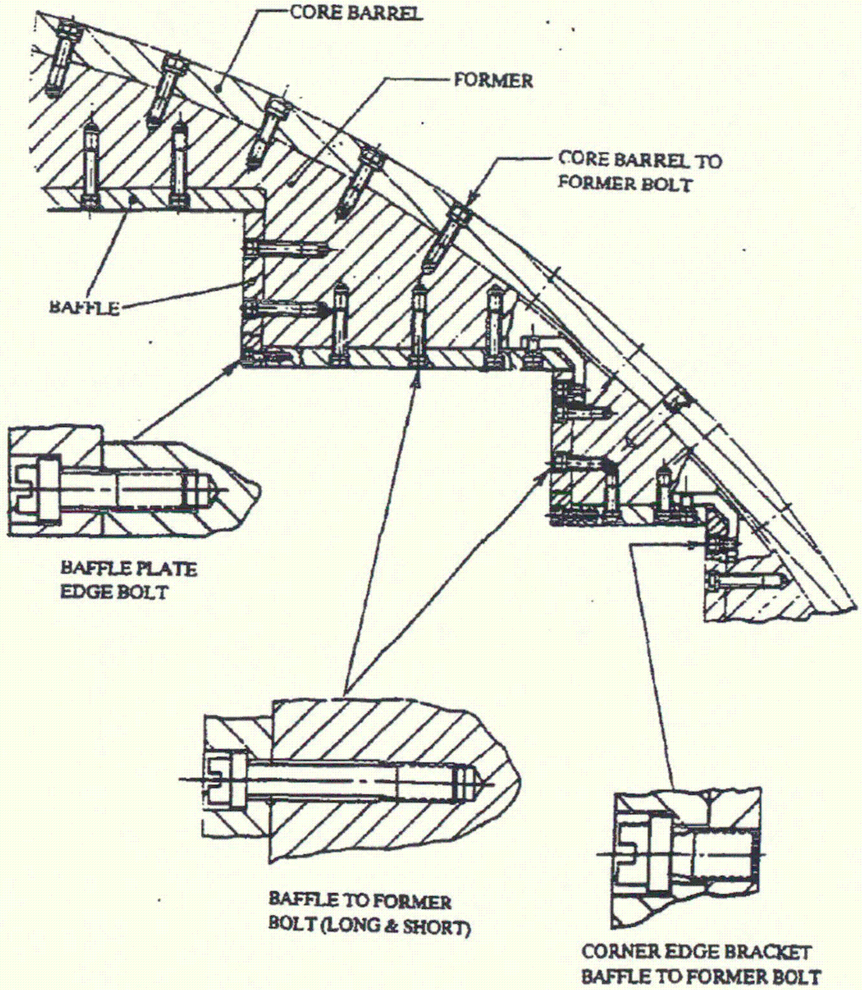

Figure A-7 Bolting in a Typical Westinghouse Baffle-Former Structure

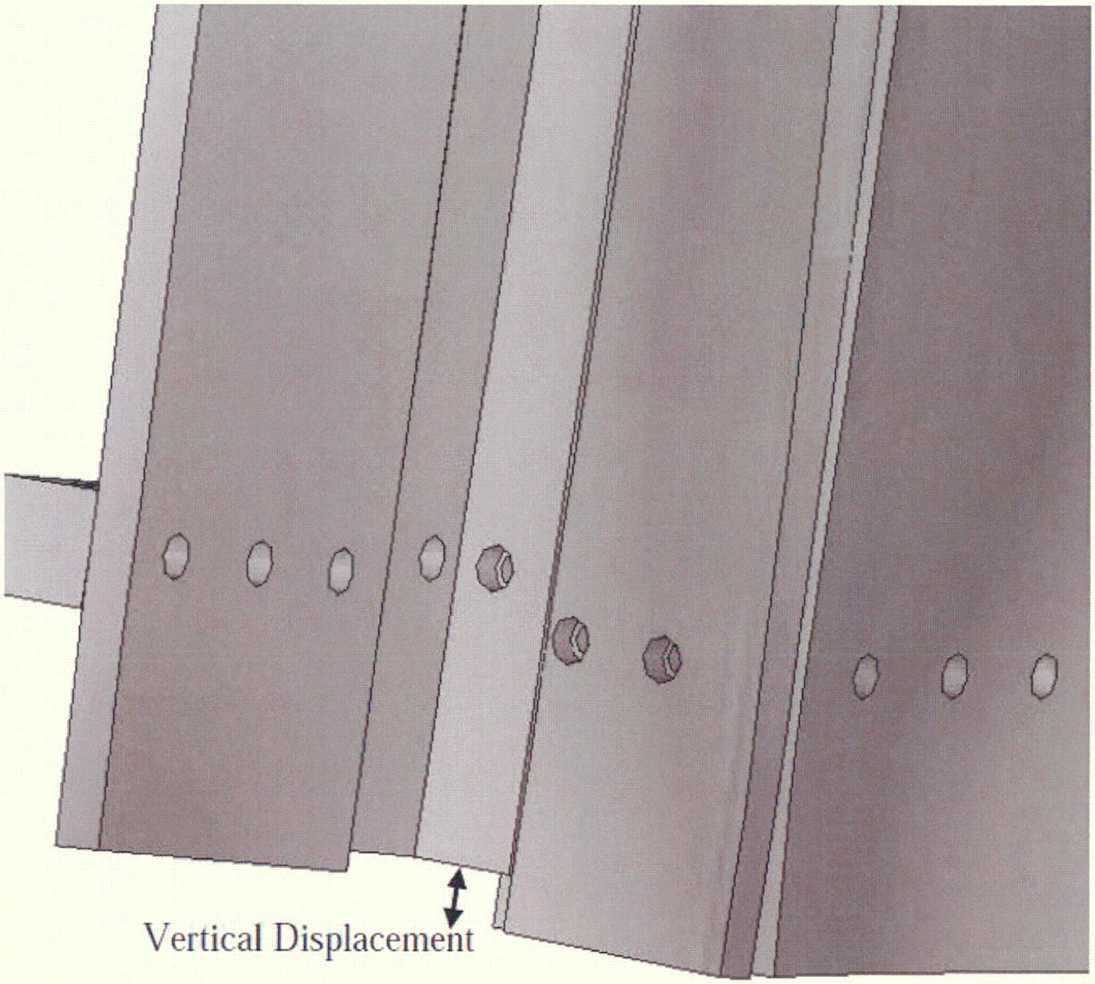

Figure A-8 Vertical Displacement between the Baffle Plates and Bracket at the Bottom of the Baffle-Former-Barrel Assembly (exaggerated)

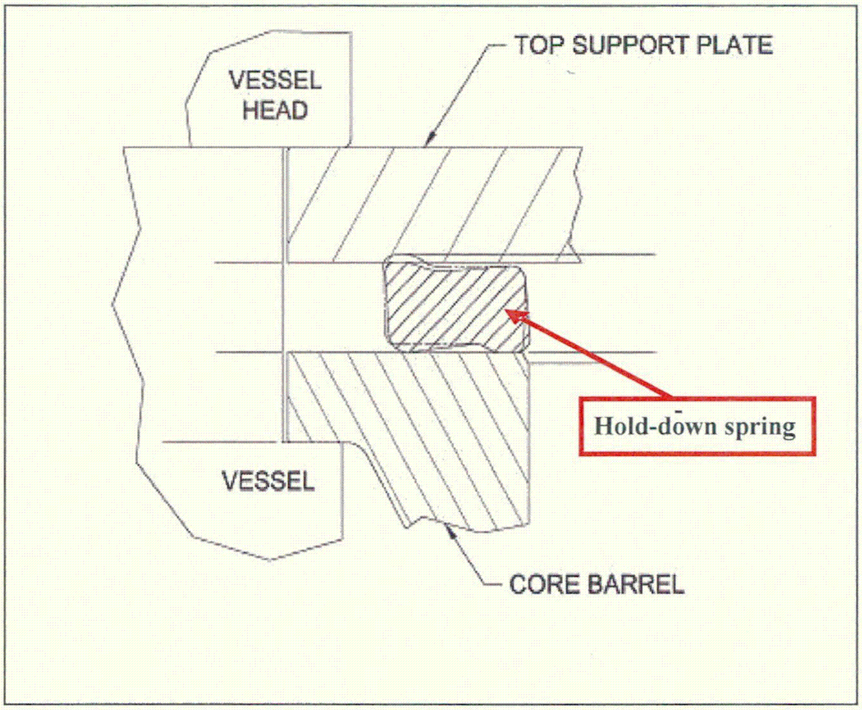

Figure A-9 Schematic Cross-Sections of the Westinghouse Hold-Down Springs

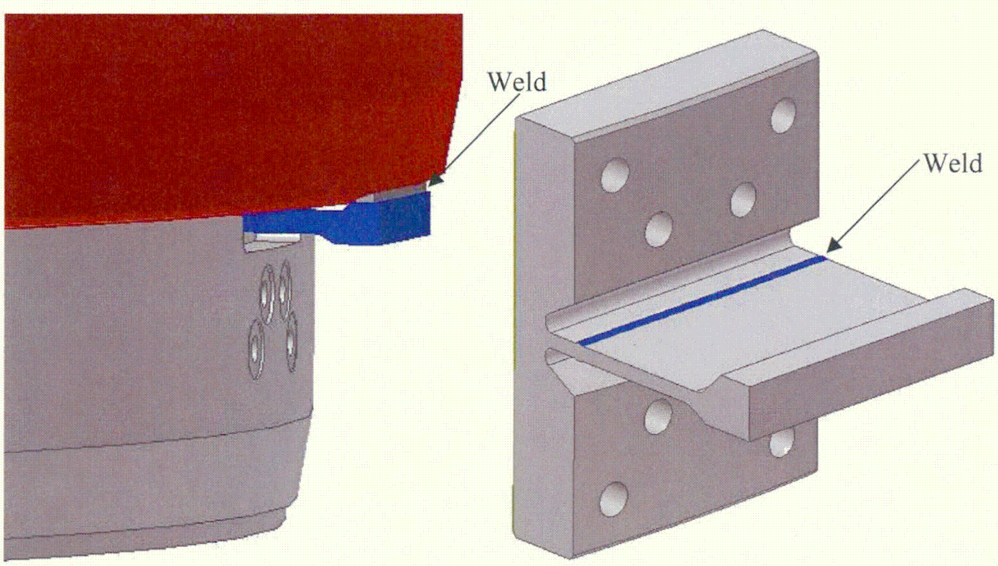

Figure A-10 Typical Thermal Shield Flexure

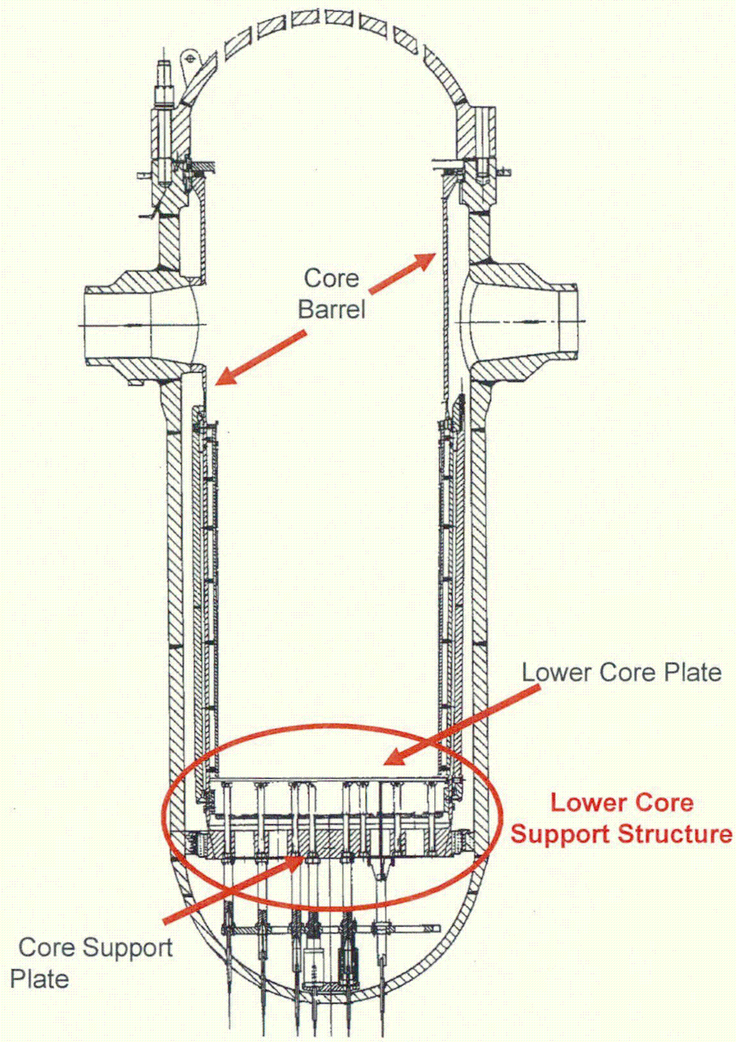

Figure A-11 Lower Core Support Structure

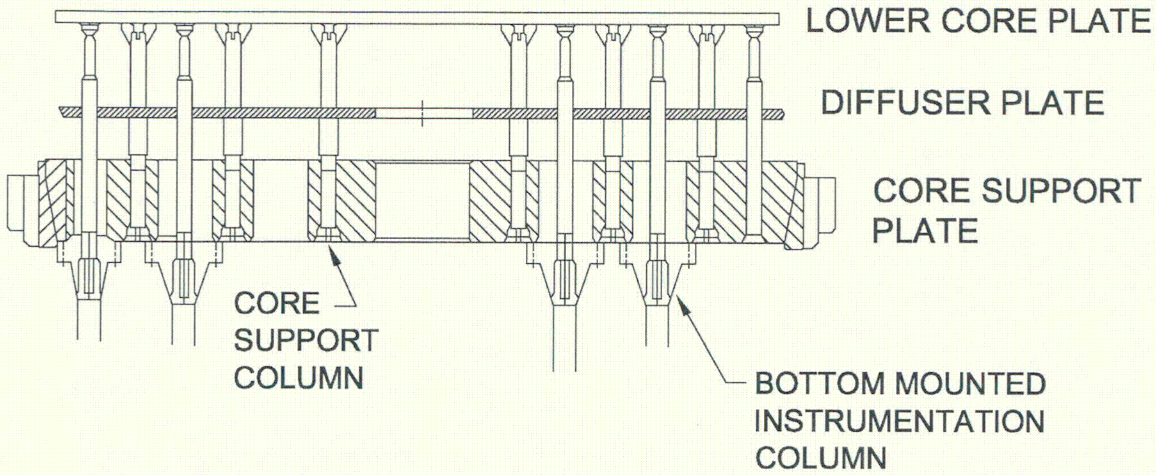

Figure A-12 Lower Core Support Structure – Core Support Plate Cross-Section

Figure A-13 Typical Core Support Column

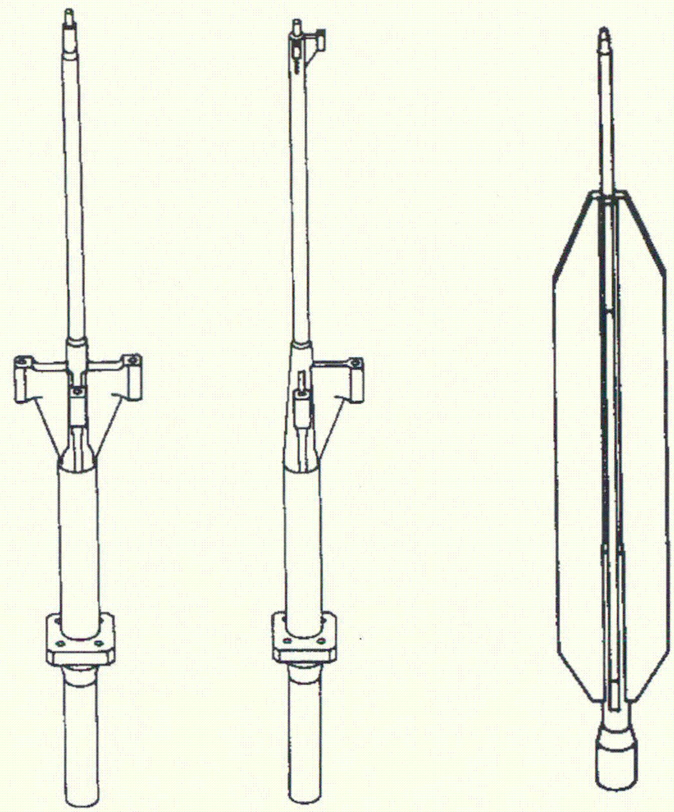

Figure A-14 Examples of Bottom-Mounted Instrumentation (BMI) Column Designs

APPENDIX B
DIABLO CANYON POWER PLANT LICENSE RENEWAL AGING MANAGEMENT REVIEW SUMMARY TABLES

The content and numerical identifiers in Table B-1 are extracted from Table 3.1.2-1 "Reactor Vessel Internals (Westinghouse) Summary of Aging Management Evaluation" of the Diablo Canyon Power Plant LRA.

Table B-1 LRA Aging Management Evaluation Summary—Table 3.1.2-1 of the DCPP LRA

Component Type	Aging Effect Requiring Management	Aging Management Program	NUREG-1801, Vol. 2 Item	LRA Table 3.1.1 Item #	Notes
RVI Control Rod Guide Tube Assembly (Guide Tube Bolts)	Loss of Material	ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD (B2.1.1)	IV.B2.BP-382	3.1.1.63	A, 3
	Cracking	ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD (B2.1.1)	IV.B2.BP-382	3.1.1.63	A, 3
RVI Core Barrel Assembly (Core Barrel Flange)	Loss of Material	ASME Section XI Inservice Inspection, Subsections IWB, IWC and IWD (B2.1.1)	IV.B2.RP-382	3.1.1.63	A, 3
RVI Core Barrel Assembly (Core Barrel Outlet Nozzles)	Cracking	ASME Section XI Inservice Inspection, Subsections IWB, IWC and IWD (B2.1.1)	IV.B2.RP-382	3.1.1.63	A, 3
	Loss of Material	ASME Section XI Inservice Inspection, Subsections IWB, IWC and IWD (B2.1.1)	IV.B2.RP-382	3.1.1.63	A, 3
RVI Core Barrel Assembly (Core Barrel Outlet Nozzle Welds)	Cracking	Water Chemistry (B2.1.2) and PWR Vessel Internals (B2.1.41)	IV.B2.RP-278	3.1.1.80	A, 3

Table B-1 LRA Aging Management Evaluation Summary—Table 3.1.2-1 of the DCPP LRA

Component Type	Aging Effect Requiring Management	Aging Management Program	NUREG-1801, Vol. 2 Item	LRA Table 3.1.1 Item #	Notes
	Loss of Fracture Toughness	PWR Vessel Internals (B2.1.41)	IV.B2.RP-278a	3.1.1.27	A, 3
RVI Core Barrel Assembly (Upper and Lower Core Barrel Girth Welds)	Cracking	Water Chemistry (B2.1.2) and PWR Vessel Internals (B2.1.41)	IV.B2.RP-387	3.1.1.30	A, 3
RVI Core Barrel Assembly (Upper and Lower Core Barrel Axial Welds)	Cracking	Water Chemistry (B2.1.2) and PWR Vessel Internals (B2.1.41)	IV.B2.RP-387a	3.1.1.80	A, 3
	Loss of Fracture Toughness	PWR Vessel Internals (B2.1.41)	IV.B2.RP-388a	3.1.1.27	A, 3
RVI Core Barrel Assembly (Lower Core Barrel Flange Weld)	Cracking	Water Chemistry (B2.1.2) and PWR Vessel Internals (B2.1.41)	IV.B2.RP-280	3.1.1.30	A, 3
RVI Core Barrel Assembly (Upper Core Barrel Flange Weld)	Cracking	Water Chemistry (B2.1.2) and PWR Vessel Internals (B2.1.41)	IV.B2.RP-276	3.1.1.30	A, 3
RVI Baffle-to-Former Assembly (Baffle Plates, Former Plates)	Changes in Dimensions	PWR Vessel Internals (B2.1.41)	IV.B2.RP-270	3.1.1.33	A, 3
	Cracking	Water Chemistry (B2.1.2) and PWR Vessel Internals (B2.1.41)	IV.B2.RP-270a	3.1.1.30	A, 3
	Loss of Material	Water Chemistry (B2.1.2)	IV.B2-32	3.1.1.83	A
RVI Baffle-to-Former Assembly (Baffle/Former Bolts)	Changes in Dimensions	PWR Vessel Internals (B2.1.41)	IV.B2.RP-272	3.1.1.33	A, 3
	Loss of Preload	PWR Vessel Internals (B2.1.41)	IV.B2.RP-272	3.1.1.33	A, 3
	Loss of Fracture Toughness	PWR Vessel Internals (B2.1.41)	IV.B2.RP-272	3.1.1.33	A, 3

Table B-1 LRA Aging Management Evaluation Summary—Table 3.1.2-1 of the DCPP LRA

Component Type	Aging Effect Requiring Management	Aging Management Program	NUREG-1801, Vol. 2 Item	LRA Table 3.1.1 Item #	Notes
RVI Baffle-to-Former Assembly (Baffle/Former Bolts)	Cracking	Water Chemistry (B2.1.2) and PWR Vessel Internals (B2.1.41)	IV.B2.RP-271	3.1.1.30	A, 3
	Loss of Material	Water Chemistry (B2.1.2)	IV.B2-32	3.1.1.83	A
RVI Baffle-to-Former Assembly (Baffle-Edge bolts)	Cracking	Water Chemistry (B2.1.2) and PWR Vessel Internals (B2.1.41)	IV.B2.RP-275	3.1.1.30	A, 3
	Loss of Fracture Toughness	PWR Vessel Internals (B2.1.41)	IV.B2.RP-354	3.1.1.33	A, 3
	Loss of Preload	PWR Vessel Internals (B2.1.41)	IV.B2.RP-354	3.1.1.33	A, 3
	Changes in Dimension	PWR Vessel Internals (B2.1.41)	IV.B2.RP-354	3.1.1.33	A, 3
RVI Lower Core Support Structure (Lower Core Plate)	Loss of Fracture Toughness	PWR Vessel Internals (B2.1.41)	IV.B2.RP-288	3.1.1.22	A, 3
	Loss of Material	PWR Vessel Internals (B2.1.41)	IV.B2.RP-288	3.1.1.22	A, 3
	Cracking	Water Chemistry (B2.1.2) and PWR Vessel Internals (B2.1.41)	IV.B2.RP-289	3.1.1.37	A, 3
RVI Lower Core Support Structure (Lower Support Forging of Casting)	Loss of Fracture Toughness	PWR Vessel Internals (B2.1.41)	IV.B2.RP-290a	3.1.1.27	A, 3
	Cracking	Water Chemistry (B2.1.2) and PWR Vessel Internals (B2.1.41)	IV.B2.RP-291a	3.1.1.80	A, 3
RVI Lower Core Support Structure	Loss of Preload	PWR Vessel Internals (B2.1.41)	IV.B2.RP-285	3.1.1.22	A, 3

Table B-1 LRA Aging Management Evaluation Summary—Table 3.1.2-1 of the DCPP LRA

Component Type	Aging Effect Requiring Management	Aging Management Program	NUREG-1801, Vol. 2 Item	LRA Table 3.1.1 Item #	Notes
(Clevis Insert Bolts)	Loss of material	PWR Vessel Internals (B2.1.41)	IV.B2.RP-285	3.1.1.22	A, 3
	Cracking	Water Chemistry (B2.1.2) and PWR Vessel Internals (B2.1.41)	IV.B2.RP-399	3.1.1.37	A, 3
RVI Lower Core Support Structure (Radial Keys, Clevis Insert Keyways)	Cumulative Fatigue Damage	Time-Limited Aging Analysis evaluated for the period of extended operation	IV.B2.RP-382	3.1.1.63	C, 3
	Cracking	ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD (B2.1.1)	IV.B2.RP-382	3.1.1.63	C, 3
	Loss of Material	ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD (B2.1.1)	IV.B2-RP-382	3.1.1.63	C, 3
RVI Lower Core Support Structure (Core Support Column Bolts)	Cracking	Water Chemistry (B2.1.2) and PWR Vessel Internals (B2.1.41)	IV.B2.RP-286	3.1.1.80	A, 3
	Loss of Fracture Toughness	PWR Vessel Internals (B2.1.41)	IV.B2.RP-287	3.1.1.27	A, 3
	Loss of Preload	PWR Vessel Internals (B2.1.41)	IV.B2.RP-287	3.1.1.27	A, 3
RVI Lower Core Support Structure (Lower Support Column Bodies [non-cast])	Cracking	Water Chemistry (B2.1.2) and PWR Vessel Internals (B2.1.41)	IV.B2.RP-294	3.1.1.80	A, 3
	Loss of Fracture Toughness	PWR Vessel Internals (B2.1.41)	IV.B2.RP-295	3.1.1.27	A, 3

Table B-1 LRA Aging Management Evaluation Summary—Table 3.1.2-1 of the DCPP LRA

Component Type	Aging Effect Requiring Management	Aging Management Program	NUREG-1801, Vol. 2 Item	LRA Table 3.1.1 Item #	Notes
RVI Lower Core Support Structure (Lower Support Column Bodies [cast])	Cracking	Water Chemistry (B2.1.2) and PWR Vessel Internals (B2.1.41)	IV.B2.RP-291	3.1.1.80	A, 3
	Loss of Fracture Toughness	PWR Vessel Internals (B2.1.41)	IV.B2.RP-290	3.1.1.27	A, 3
RVI Lower Core Support Structure (All RVI Stainless Steel Components)	Cumulative Fatigue Damage	Time-Limited Aging Analysis evaluated for the period of extended operation	IV.B2-31	3.1.1.05	A
	Loss of Material	Water Chemistry (B2.1.2)	IV.B2-32	3.1.1.83	A
RVI Lower Core Support Structure (Core Support Casting [U1])	Cumulative Fatigue Damage	Time-Limited Aging Analysis evaluated for the period of extended operation	IV.B2-31	3.1.1.05	A
	Loss of Material	Water Chemistry (B2.1.2)	IV.B2-32	3.1.1.83	A
RVI Thermal Shield Assembly (Thermal Shield Flexures)	Loss of Material	PWR Vessel Internals (B2.1.41)	IV.B2.RP-302a	3.1.1.33	A, 3
	Cracking	Water Chemistry (B2.1.2) and PWR Vessel Internals (B2.1.41)	IV.B2.RP-302	3.1.1.30	A, 3
RVI Upper Core Support Structure (Upper Core Plate)	Cumulative Fatigue Damage	Time-Limited Aging Analysis evaluated for the period of extended operation	IV.B2-31	3.1.1.05	A
	Loss of Material	PWR Vessel Internals (B2.1.41)	IV.B2.RP-290b	3.1.1.27	A, 3
	Cracking	PWR Vessel Internals (B2.1.41)	IV.B2.RP-291b	3.1.1.80	A, 3

Table B-1 LRA Aging Management Evaluation Summary—Table 3.1.2-1 of the DCPP LRA

Component Type	Aging Effect Requiring Management	Aging Management Program	NUREG-1801, Vol. 2 Item	LRA Table 3.1.1 Item #	Notes
RVI Upper Core Support Structure (Upper Core Plate Alignment Pins)	Loss of Material	PWR Vessel Internals (B2.1.41)	IV.B2.RP-299	3.1.1.22	A, 3
	Cracking	Water Chemistry (B2.1.2) and PWR Vessel Internals (B2.1.41)	IV.B2.RP-301	3.1.1.37	A, 3
RVI Upper Core Support Structure (Upper Support Columns)	Cumulative Fatigue Damage	Time-Limited Aging Analysis evaluated for the period of extended operation	IV.B2-31	3.1.1.05	A
	Loss of Material	Water Chemistry (B2.1.2)	IV.B2-32	3.1.1.83	A
RVI Upper Core Support Structure (Upper Support Ring or Skirt)	Cracking	Water Chemistry (B2.1.2) and PWR Vessel Internals (B2.1.41)	IV.B2.RP-346	3.1.1.37	A, 3
RVI Hold-Down Spring (RVI Hold-Down Spring)	Loss of Material	Water Chemistry (B2.1.2)	IV.B2.RP-300	3.1.1.33	A, 3
	Loss of Preload	PWR Vessel Internals (B2.1.41)	IV.B2.RP-300	3.1.1.33	A, 3
	Changes in Dimensions	PWR Vessel Internals (B2.1.41)	IV.B2.RP-300	3.1.1.33	A, 3
RVI Control Rod Guide Tube Assembly (Control Rod Guide Tubes/Tube Support Pins/Guide Plates [Cards])	Loss of Material	PWR Vessel Internals (B2.1.41)	IV.B2.RP-296	3.1.1.33	A, 3
RVI Control Rod Guide Tube Assembly (Lower Flange Welds)	Loss of Fracture Toughness	PWR Vessel Internals (B2.1.41)	IV.B2.RP-297	3.1.1.33	A, 3

Table B-1 LRA Aging Management Evaluation Summary—Table 3.1.2-1 of the DCPP LRA

Component Type	Aging Effect Requiring Management	Aging Management Program	NUREG-1801, Vol. 2 Item	LRA Table 3.1.1 Item #	Notes
RVI Components	Loss of Material	Water Chemistry (B2.1.2)	IV.B2-32	3.1.1.83	A
RVI Control Rod Guide Tube Assembly (Guide Tube Bolts)	Cracking	ASME Section XI Inservice Inspection, Subsections IWB, IWC and IWD (B2.1.1)	IV.B2.BP-382	3.1.1.63	A, 3
	Loss of Material	ASME Section XI Inservice Inspection, Subsections IWB, IWC and IWD (B2.1.1)	IV.B2.BP-382	3.1.1.63	A, 3
RVI Baffle-to-Former Assembly (Barrel-to-Former Bolts)	Changes in Dimensions	PWR Vessel Internals (B2.1.41)	IV.B2.RP-274	3.1.1.27	A, 3
	Loss of Preload	PWR Vessel Internals (B2.1.41)	IV.B2.RP-274	3.1.1.27	A, 3
	Cracking	Water Chemistry (B2.1.2) and PWR Vessel Internals (B2.1.41)	IV.B2.RP-273	3.1.1.80	A, 3
	Loss of Fracture Toughness	PWR Vessel Internals (B2.1.41)	IV.B2.RP-274	3.1.1.27	A, 3
BMI System (BMI Column Bodies)	Loss of Fracture Toughness	PWR Vessel Internals (B2.1.41)	IV.B2.RP-292	3.1.1.27	A, 3
	Cracking	Water Chemistry (B2.1.2) and PWR Vessel Internals	IV.B2.RP-293	3.1.1.80	A, 3

Standard Notes:

A Consistent with NUREG-1801 item for component, material, environment, and aging effects. AMP is consistent with NUREG-1801 AMP.

C Component is different, but consistent with NUREG-1801 item for material, environment, and aging effect. AMP is consistent with NUREG-1801 AMP.

E Consistent with NUREG-1801 for material, environment, and aging effect, but a different aging management program is credited or NUREG-1801 identifies a plant-specific aging management program.

Plant-Specific Note:

3 Line item was revised to align with NUREG-1801, Revision 2 [6] and LR-ISG-2011-04. Reference PG&E Letter DCL-14-103 [34]

APPENDIX C
MRP-227 AUGMENTED INSPECTIONS

Table C-1 MRP-227 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(7). (Reference 42)	Minimum examination of 20% of the number of CRGT assemblies, and as per the requirements of WCAP-17451-P Revision 1, Section 5(7). See Figure A-2. (Reference 42)
Control Rod Guide Tube Assembly Lower flange welds	All plants	Cracking (SCC, Fatigue) Aging Management (IE and TE)	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 two refueling outages from the beginning of the license renewal period and subsequent examination on a 10-year interval.	100% of outer (accessible) CRGT lower flange weld surfaces and adjacent base metal on the periphery CRGT assemblies(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 10-year interval.	100% of one side of the accessible surfaces of the selected weld and adjacent base metal(4). See Figure A-4.

Table C-1 MRP-227 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
Core Barrel Assembly Upper and lower core barrel cylinder girth welds	All plants	Cracking (SCC, IASCC, Fatigue)	None	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(4).
Core Barrel Assembly Lower core barrel flange weld(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 examination on a ten-year interval.	100% of one side of the accessible surfaces of the selected weld and adjacent base metal(4).

Table C-1 MRP-227 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-edge bolts	All plants with baffle-edge bolts (Applicable to DCPP 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)[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[3]. See Figures A-5, A-6, and A-7.
Baffle-Former Assembly Baffle-former bolts	All plants	Cracking (IASCC, Fatigue)	Lower support column bolts, barrel-former bolts	Baseline volumetric (UT) examination between 25 and 35 EFPY, with subsequent examination after 10 to 15 additional EFPY to confirm stability of bolting pattern. Re-examination for high-leakage core designs requires continuing examinations on a ten-year interval.	100% of accessible bolts[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-6.

Table C-1 MRP-227 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 Assembly	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 joints	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 Figure A-5 and A-8.

Table C-1 MRP-227 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
Alignment and Interfacing Components Internals hold down spring	All plants with 304 stainless steel hold down springs (Applicable to DCPP Unit 1)	Distortion (Loss of Load) Note: This mechanism was not strictly identified in the original list of age-related degradation mechanisms (ARDM).	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. Replacement of 304 springs by 403 springs is required when the spring stiffness is determined to relax beyond design tolerance. See Figure A-9.
Thermal Shield Assembly Thermal shield flexures	All plants with thermal shields (Applicable to DCPP 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 MRP-227 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

Note:
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 Table 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 Table 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 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.

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

Item	Applicability	Effect (Mechanism)	Primary Link (Note 1)	Examination Method (Note 1)	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(2).
Lower Internals Assembly Lower Support Forging or Castings	All plants	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(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(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(2). See Figures A-11, A-12 and A-13.
Core Barrel Assembly Core barrel outlet nozzles	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(2). See Figure A-4.

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

Item	Applicability	Effect (Mechanism)	Primary Link (Note 1)	Examination Method (Note 1)	Examination Coverage
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(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(2). See Figures A-11, 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(2). See Figures A-11, A-12, and A-13.
BMI System 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. Flux thimble insertion/withdrawal to be monitored at each inspection interval. Re-inspection every 10 years following initial inspection.	100% of BMI column bodies for which difficulty is detected during flux thimble insertion/withdrawal See Figures A-12 and A-14.

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

Item	Applicability	Effect (Mechanism)	Primary Link (Note 1)	Examination Method (Note 1)	Examination Coverage

Note:
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, 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).

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

Item	Applicability	Effect (Mechanism)	Primary Link	Examination Method/Frequency	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[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[1]	All plants	Loss of material (Wear)	ASME Code Section XI	Visual (VT-3) examination.	All accessible surfaces at specified frequency.
BMI 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)[2]	ASME Code Section XI	Visual (VT-3) examination.	All accessible surfaces at specified frequency.

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

Item	Applicability	Effect (Mechanism)	Primary Link	Examination Method/Frequency	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, which is not applicable at DCPP Unit 1.
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 Acceptance Criteria and Expansion Criteria Recommendations for Westinghouse-Designed Internals

Item	Applicability	Examination Acceptance Criteria(1)	Expansion Link(s)	Expansion Criteria	Additional Examination Acceptance Criteria
Control Rod Guide Tube Assembly Guide plates (cards)	All plants	Visual (VT-3) Examination(2) 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
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. 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 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 MRP-227 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
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, and any supplementary UT examination, be expanded to include the core outlet nozzle welds by the completion of the next refueling outage. b. If extensive cracking in the remaining core barrel 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 is a detectable crack-like surface indication.

Table C-4 MRP-227 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
Core Barrel Assembly Lower core barrel flange weld(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 2 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 MRP-227 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
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 2 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.
Baffle-Former Assembly Baffle-edge bolt	All plants with baffle-edge bolts (Applicable to DCPP 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 MRP-227 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
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 MRP-227 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
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 MRP-227 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
Alignment and Interfacing Components Internals hold down spring	All plants with 304 stainless steel hold down springs (Applicable to DCPP Unit 1)	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
Thermal Shield Assembly Thermal shield flexures	All plants with thermal shields (Applicable to DCPP 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

Table C-4 MRP-227 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

Note:
1. The examination acceptance criterion for visual examination is the absence of the specified relevant condition(s).
2. WCAP-17451-P Revision 1 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.

Enclosure 4
PG&E Letter DCL-15-150

WCAP-17463-NP, Revision 1

Program Plan for Aging Management of Reactor Vessel Internals at Diablo Canyon Power Plant Unit 2

WCAP-17463-NP
Revision 1

December 2015

Program Plan for Aging Management of Reactor Vessel Internals at Diablo Canyon Power Plant Unit 2

Westinghouse

WCAP-17463-NP
Revision 1

Program Plan for Aging Management of Reactor Vessel Internals at Diablo Canyon Power Plant Unit 2

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LIST OF ACRONYMS

3-D	three-dimensional
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
CRGT	control rod guide tube
CUF	cumulative usage factor
DCPP	Diablo Canyon Power Plant
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 mode, effects, and criticality analysis
GALL	Generic Aging Lessons Learned
I&E	inspection and evaluation
IASCC	irradiation-assisted stress corrosion cracking
IE	irradiation embrittlement
IGSCC	intergranular stress corrosion cracking
INPO	Institute of Nuclear Power Operations
ISI	inservice inspection
IP	industry program
ISR	irradiation-enhanced stress relaxation
LRA	License Renewal Application
MRP	Materials Reliability Program
NDE	nondestructive examination
NEI	Nuclear Energy Institute
NOS	Nuclear Oversight Section
NRC	U.S. Nuclear Regulatory Commission
NSSS	nuclear steam supply system
OE	operating experience
OER	Operating Experience Report
PH	precipitation-hardenable
PG&E	Pacific Gas and Electric Company
PWR	pressurized water reactor
PWROG	Pressurized Water Reactor Owners Group (formerly WOG)

LIST OF ACRONYMS (cont.)

PWSCC	primary water stress corrosion cracking
QA	quality assurance
RCS	reactor coolant system
RFO	refueling outage
RV	reactor vessel
RVI	reactor vessel internals
SCC	stress corrosion cracking
SE	safety evaluation
SER	Safety Evaluation Report
SRP	Standard Review Plan
SS	stainless steel
TE	thermal embrittlement
UFSAR	Updated Final Safety Analysis Report
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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1 PURPOSE

The purpose of this report is to document the Diablo Canyon Power Plant (DCPP) Unit 2, hereafter DCPP Unit 2, Reactor Vessel Internals (RVI) Aging Management Program Plan (AMP). The purpose of the AMP is to manage the effects of aging on reactor vessel internals through the license renewal period, which for DCPP Unit 2 begins at midnight on August 26, 2025. This document provides a description of the program as it relates to the management of aging effects identified in various regulatory, industry-generated, and DCPP Unit 2 plant-specific documents in support of license renewal program evaluations. This AMP is prepared in accordance with various regulatory and industry-generated documents, and is supported by existing DCPP Unit 2 documents and procedures. As required, by industry experience or directive in the future, the DCPP Unit 2 RVI AMP will be updated or supported by additional documents to provide clear and concise direction for the effective management of aging degradation in the DCPP Unit 2 reactor internals components. These actions provide assurance that operations at DCPP Unit 2 will continue to be conducted in accordance with the current licensing basis (CLB) for the reactor vessel internals by fulfilling license renewal commitments (Reference 1), American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel (B&PV) Code Section XI Inservice Inspection (ISI) requirements (Reference 2), and industry requirements (Reference 3). 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 DCPP Unit 2 RVI AMP are to:

- Demonstrate that the effects of aging on the RVI will be adequately managed for the period of extended operation in accordance with 10 CFR 54 (Reference 4).
- Summarize the role of existing DCPP Unit 2 AMPs in the RVI AMP.
- Define and implement the industry-defined (EPRI/MRP and PWROG) pressurized water reactor (PWR) RVI requirements and guidance for managing aging of reactor internals.
- Provide an inspection plan summary for the DCPP Unit 2 reactor internals.

DCPP License Renewal Commitments 72 & 73 (Reference 34) defines the content and timeline for the program that Pacific Gas and Electric Company (PG&E) has committed to implement for the reactor vessel internals components:

> *Commitment 72*: Prior to the period of extended operation, implement the PWR Vessel Internals Program to conform to LR-ISG-2011-04 as discussed in PG&E Letter DCL-14-103, Enclosure 1, Attachment 4, including the plant-specific action items, conditions, and limitations identified in the NRC Safety Evaluation, Revision 1, for MRP-227.
>
> *Commitment 73*: The NRC SE for MRP-227 contains eight action items for applicants/licensees to consider. Responses to the applicable aging management program plant-specific action items, conditions and limitations identified in the NRC SE, Revision 1, on MRP-227 will be submitted to the NRC by December 2015. Reference DCP-14-103, Enclosure 1, Attachment 4.

Augmented inspections, based on required program enhancements resulting from ongoing and future industry programs, will be incorporated into the DCPP Unit 2 RVI AMP. The Program Plan for Inspection of Reactor Vessel Internals will coordinate with the existing DCPP Unit 2 Inservice Inspection Program (Reference 2), and will supplement that program with the augmented examinations for managing the potential aging effects of the reactor vessel internals. Corrective actions for augmented inspections will either be developed using repair and replacement procedures equivalent to those requirements in ASME B&PV Code, Section XI (Reference 5), or equivalent or more rigorous procedures will be determined by PG&E independently or in cooperation with the industry. PG&E is currently committed to the 2001 Edition through the 2003 Addenda of the ASME Code for DCPP Unit 2, and initial development of this AMP will be based on this edition; however, for future development, later editions and addenda will be invoked as required by 10 CFR 50.55a or approved NRC Code Cases or Safety Evaluation Reports (SERs).

This AMP for the DCPP Unit 2 reactor internals demonstrates that the Reactor Vessel Internals Program adequately manages the effects of aging for reactor internals components. The AMP also establishes the basis for providing reasonable assurance that the internals components will continue to perform their intended function through the DCPP Unit 2 license renewal period of extended operation. It also supports the DCPP Unit 2 License Renewal Commitment 72 to implement the PWR Vessel Internals Program to conform to LR-ISG-2011-04 as discussed in PG&E Letter DCL-14-103, Enclosure 1, Attachment 4, including the plant-specific action items, conditions and limitations identified in the NRC Safety Evaluation, Revision 1, for MRP-227. This AMP also supports the DCPP Unit 2 License Renewal Commitment 73, which is to provide responses to the applicable aging management program plant-specific action items, conditions and limitations identified in the NRC SE, Revision 1, on MRP-227 to the NRC by December 2015. Furthermore, this AMP will demonstrate the consistency of the program with that documented in NUREG-1801, December 2010, Section XI.M16A (Reference 6). The development of this program satisfies the MRP-227 requirements.

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) (Reference 7). In recent years, the U.S. nuclear power industry has been actively engaged 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 (WOG) issued WCAP-14577 (Reference 8), "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, the EPRI MRP engaged in an effort 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, and on the accumulated industry research data, the following elements of an Aging Management Program were further developed (References 8, 9, and 10):

- 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. These categories ranged from components for which the effects from the postulated aging mechanisms are insignificant, to components that are moderately susceptible to the aging effects, to 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 for implementing the appropriate aging management methodology, baseline examination timing, and the need and timing of subsequent inspections were developed. Development of these strategies was based on combining the results of functionality assessment with several contributing factors, including component accessibility, operating experience, existing evaluations, and prior examination results.

The industry effort, as coordinated by the EPRI MRP, has finalized initial Inspection and Evaluation (I&E) Guidelines for reactor internals and submitted the document to the NRC with a request for a formal SER. A supporting document addressing inspection requirements has also been completed. The industry guidance is contained within two separate EPRI MRP documents:

- MRP-227-A, "PWR Internals Inspection and Evaluation Guidelines" (Reference 3) (hereafter referred to as "the I&E Guidelines" or simply "MRP-227-A") provides the industry background, listing of reactor internals components requiring inspection, type of non-destructive evaluation (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, "Inspection Standard for PWR Internals" (Reference 11), provides guidance on the qualification and demonstration of the NDE techniques and on other criteria pertaining to the actual performance of the inspections.

The PWROG has also developed "Reactor Internals Acceptance Criteria Methodology and Data Requirements" for the MRP-227 inspections, where feasible (Reference 12). Final reports are 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 if a generic approach is not practical.

The DCPP reactor internals for Unit 2 are integral with the reactor coolant system (RCS) of a Westinghouse four-loop nuclear steam supply system (NSSS). A typical illustration of the reactor vessel internals is provided in Figure A-1.

The DCPP License Renewal Application (LRA), Section 2.3.1, (Reference 13) describes the DCPP Unit 2 reactor vessel internals. The RVI consist of the lower core support structure (including the entire core barrel and the neutron pad assembly), the upper core support structure, and the incore instrumentation support structures. The reactor internals support the core, maintain fuel alignment, limit fuel assembly movement, maintain alignment between fuel assemblies and CRDMs, direct coolant flow to the pressure vessel head, provide gamma and neutron shielding, and provide guides for incore instrumentation.

The lower core support structure includes the baffle and former plates, core barrel assembly, neutron shield panels, lower core plates, core support forging, support columns, secondary core support, energy absorbers, tie plates, manway cover, and support ring.

The upper core support structure includes the upper support columns, upper support plate, upper core plate, and control rod guide tubes. DCPP Unit 2 features head cooling spray nozzles that are holes machined in the upper support plate flange.

The incore instrumentation support structure includes the flux thimble tubes and guide tubes, seal table and fittings, and upper instrumentation columns. Components that provide interfaces between the major assemblies include the radial keys, clevis inserts, fuel alignment pins, head/vessel alignment pins, upper core plate alignment pins, and hold-down spring.

The NRC has issued an SER to Diablo Canyon that states that the NRC will not finalize a decision on license renewal until completion of three-dimensional (3-D) seismic studies and PG&E's receipt of a coastal consistency certification. The staff will supplement this SER, as necessary, considering any relevant new information from the seismic studies, operating experience, and annual updates prior to finalizing a decision on license renewal. In the SER, the NRC concluded that the DCPP LRA (Reference 13) adequately identified the RVI systems, structures, and components that are subject to an aging management review (AMR), and that the requirements of 10 CFR 54.29(a) (Reference 4) had been met. Appendix B, Table B-1 of this report lists the DCPP Unit 2 reactor vessel internals components and subcomponents subject to AMR requirements according to Table 3.1.2-2 of the DCPP LRA.

The U.S. nuclear industry, as noted 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) protocol NEI 03-08 (Reference 14), "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. MRP-227 was issued in December 2008, and plant AMPs must have been completed by December 2011, or sooner, if required by plant-specific license renewal commitments. Per the MRP-227 requirement, DCPP Unit 2 completed development of the AMP by the due date of December 2011 via the issuance of Revision 0 of this WCAP.

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 current licensing basis for the period of extended operation.

3 PROGRAM OWNER

The successful implementation and comprehensive long-term management of the DCPP Unit 2 RVI AMP will require PG&E to interact with multiple industry organizations including, but not limited to, the ASME, MRP, NRC, and PWROG. The responsibilities of the individual PG&E organizational groups are provided in the following paragraphs. PG&E will maintain cognizance of industry activities related to PWR internals inspection and aging management and will address and implement industry guidance stemming from those activities, as appropriate under NEI 03-08 practices.

The overall responsibility for administration of the RVI AMP lies with PG&E Senior Management.

Additional responsibilities and the appropriate responsible personnel are discussed in the following subsections.

3.1 ENGINEERING DIRECTOR

- Approves implementation of the RVI AMP.
- Ensures coordination and implementation of the RVI AMP.

3.2 RCS MATERIALS DEGRADATION MANAGEMENT PROGRAM OWNER

- Acts as the point of contact with materials-related industry programs (IPs) and NEI
- Ensures new requirements or recommendations issued under NEI 03-08 are disseminated to the appropriate program owners or responsible personnel.
- Initiates Notifications in the Corrective Action Program for tracking of NEI 03-08 Mandatory and Needed requirements.
- Ensures timely reporting and management of new or unexpected RCS material issues in accordance with the emergent issue protocol of NEI 03-08.
- Processes any deviations taken from IP guidelines in accordance with NEI 03-08 requirements.

3.3 REACTOR VESSEL INTERNALS PROGRAM MANAGER

- Overall development of the RVI AMP.
- Administers and oversees implementation of the RVI AMP.
- Ensures that regulatory requirements related to inspection activities are met and incorporated into the RVI AMP.
- Communicates with senior management on relevant industry experience.

- Maintains the RVI AMP to incorporate changes and updates based on industry operating experience and benchmarking results.

- Ensures prompt notification of the RCS Materials Degradation Management Program Owner whenever an issue or indication of potential generic Industry significance is identified.

- Participates in the planning and implementation of inspections of the RVI.

- Participates in industry programs related to RVI aging management.

3.4 INSERVICE INSPECTION ENGINEER

- Plans and implements inspections required by ASME Section XI B-N-3, the supplemental inspections identified in this RVI AMP, and any other plant-specific commitments for inspections related to managing the aging of RVI.

- Reviews, evaluates, and dispositions inspection results.

- Reviews and approves vendor NDE procedures and personnel qualifications.

- Provides direction and oversight of contracted NDE activities.

3.5 WORK CONTROL MANAGER

- Integrates required activities into the appropriate outage plans.

3.6 CHEMISTRY MANAGER

- Maintains primary water chemistry in accordance with approved DCPP procedures and specifications.

- Ensures that DCPP documentation supports and incorporates the guidance of industry programs including but not limited to the EPRI Primary Water Chemistry Guidelines (Reference 9).

4 DESCRIPTION OF THE DIABLO CANYON POWER PLANT UNIT 2 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 aging management programs based on inspection and evaluation. The intent of the DCPP Unit 2 AMP is to ensure the long-term integrity and safe operation of the reactor internals components. PG&E has developed this AMP in conformance with the 10 Generic Aging Lessons Learned (GALL) (Reference 6) attributes and MRP-227-A (Reference 3).

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 (References 9 and 15), inspections prescribed by the "Inservice Inspection Program Implementation" (Reference 2), thimble tube inspections (Reference 17), and past and future mitigation projects such as control rod guide tube support pin replacement, combined with augmented inspections or evaluations as recommended by MRP-227-A.

Aging degradation mechanisms that affect internals have been identified for DCPP Unit 2 and are documented in the LRA submitted for DCPP (Reference 13). 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 DCPP Unit 2 AMR methodology 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. The actual mechanism that causes SCC involves a complex interaction of environmental and metallurgical factors. The aging effect is cracking.

- <u>Primary Water Stress Corrosion Cracking</u>

  Primary water stress corrosion cracking (PWSCC) is a unique form of SCC that occurs as a result of the chemistry of primary coolant acting on primary components fabricated from susceptible materials. 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.

- Wear (loss of material)

  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 (cracking)

  Fatigue is the structural deterioration that can occur as the result of repeated stress/strain cycles caused by fluctuating loads and/or 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 (reduction in fracture toughness)

  Thermal aging embrittlement is the exposure of delta ferrite within cast austenitic stainless steel (CASS) and precipitation-hardenable (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 (reduction in fracture toughness)

  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 (distortion)

  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 incore 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 (loss of preload, or loss of mechanical closure integrity)

  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 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 at PWR internals temperatures.

  Creep (or more precisely, secondary creep) is a slow, time- and temperature-dependent, plastic deformation of materials that can occur at 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 considering 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 DCPP 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 (Reference 6). In the DCPP Unit 2 RVI AMP, this is demonstrated through application of the existing DCPP Unit 2 AMR methodology that credits inspections prescribed by the ASME Section XI Inservice Inspection Program, which will become part of the DCPP Unit 2 Inspection Program Plan for Reactor Vessel Internal along with existing DCPP Unit 2 programs, and additional augmented inspections based on MRP-227-A recommendations. The Program Plan for Inspection of Reactor Vessel Internals will coordinate with the existing DCPP Unit 2 Inservice Inspection Program (Reference 2) and will supplement that program with the augmented examinations for managing the potential aging effects of the reactor vessel internals. A description of the applicable existing DCPP Unit 2 programs and compliance with the elements of the GALL is contained in the following subsections.

4.1 EXISTING DIABLO CANYON POWER PLANT UNIT 2 PROGRAMS

PG&E's overall strategy for managing aging in reactor internals components at DCPP Unit 2 is supported by the following existing programs:

- Water Chemistry
- ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD
- Flux Thimble Tube Inspection

These are established programs that support the aging management of RCS components in addition to the RVI components. Although affiliated with and supporting the RVI AMP, they will be managed under the existing programs.

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

4.1.1 Water Chemistry

The DCPP Unit 2 Primary Strategic Water Chemistry Plan (Reference 15) is used to mitigate aging effects on component surfaces that are exposed to PWR primary water as process fluid. Chemistry programs are used to control water chemistry for impurities that accelerate corrosion and contaminants that may cause cracking due to SCC. This program relies on monitoring and control of water chemistry to keep operating levels of various contaminants below the system-specific limits. The Primary Strategic Water Chemistry Plan is based on the EPRI PWR Primary Water Chemistry Guidelines (Reference 9). The limits imposed by the DCPP Unit 2 program meet the intent of the industry standard for addressing primary water chemistry (Reference 9).

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

4.1.2 ASME Section XI Inservice Inspection, Subsections IWB, IWC and IWD

The DCPP Unit 2 Inservice Inspection Program Implementation (Reference 2) is used to monitor for aging effects such as cracking, loss of preload due to stress relaxation or irradiation creep, loss of material, and reduction of fracture toughness due to thermal embrittlement. For DCPP Unit 2, inspections conducted under the reactor internals AMPs will be controlled as a combination of ASME Section XI ISI exams on core support structures and augmented exams performed under that ISI Program, which will become part of the DCPP Unit 2 Inspection Program Plan for Reactor Vessel Internals for the remaining reactor internals components addressed within MRP-227-A. The Program Plan for Inspection of Reactor Vessel Internals will coordinate with the existing DCPP Unit 2 Inservice Inspection Program (Reference 2) and will supplement that program with the augmented examinations for managing the potential aging effects of the reactor vessel internals. The DCPP Unit 2 Section XI, 10-year ISI examination supporting the license renewal period is scheduled to take place during Spring 2026, 2R25. This is based on average 20-month cycles. The subsequent 10-year ISI will take place during Cycle 31.

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

4.1.3 Flux Thimble Tube Inspection Program

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 and 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. In turn, the flux thimbles provide paths for the neutron flux detectors into the core and are subject to reactor coolant pressure on the outside and containment pressure on the inside.

The DCPP Unit 2 Flux Thimble Tube Inspection program is an existing plant-specific program that satisfies NRC Bulletin 88-09 (Reference 19) requirements that a tube wear inspection procedure (References 17 and 18) be established and maintained for Westinghouse-supplied reactors that use bottom-mounted flux thimble tube instrumentation. Details of the program are given in the DCPP LRA, Section B2.1.21, page B-93. The program includes eddy current testing requirements for thimble tubes and criteria for determining sample size (includes all thimble tubes installed in the reactor vessel), inspection frequency, flaw evaluation, and corrective actions. The Flux Thimble Tube Inspection Program effectively manages aging effects by identifying loss of material due to wear in the thimble tubes prior to leakage. Continued implementation of this program provides reasonable assurance that aging effects will be managed such that the BMI thimble tubes will continue to perform their intended function consistent with the current licensing basis for the period of extended operation.

The SER for the DCPP LRA (Reference 1) reviewed the Flux Thimble Tube Inspection program against the 10 program elements of the GALL (Reference 6) for Program XI.M37 and determined that the aging effects would be adequately managed for the period of extended operation. The SER evaluation remains applicable.

4.2 SUPPORTING DIABLO CANYON POWER PLANT UNIT 2 PROGRAMS AND AGING MANAGEMENT SUPPORTIVE PLANT ENHANCEMENTS

4.2.1 Reactor Internals Aging Management Review Process

A comprehensive review of aging management was performed for the DCPP Unit 2 reactor vessel internals components according to the requirements of the License Renewal Rule (Reference 4). This review was conducted in support of the DCPP Unit 2 license renewal for reactor internals (Reference 33). The DCPP LRA (Reference 13) was approved by the NRC in Reference 1. Subsection 2.3.1.1 and Table 2.3.1-1 of the LRA identified the components that are subject to AMR. Table 3.1.2-1 of the LRA provides the detailed results of the AMRs conducted on these components and includes a comparison to NUREG-1801, Volume 2 to note consistencies. Appendix B, Table B-1 of the DCPP Unit 2 AMP includes a portion of LRA Table 3.1.2-1.

The aging management review supported the 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 to augment existing programs or to create new aging management programs if the existing programs were found to be inadequate to manage the aging effects

Aging management reviews were performed for each DCPP Unit 2 system that contained long-lived, passive components requiring aging management review, and the results are incorporated into the DCPP LRA.

4.2.2 Flux Thimble Tubes

The Flux Thimble Tube Inspection program manages loss of material by performing wall thickness eddy current testing of all flux thimble tubes that form part of the RCS pressure boundary. The pressure boundary includes the length of the tube inside the reactor vessel out to the seal fittings outside the reactor vessel. Eddy current testing is performed on the portion of the tubes inside the reactor vessel. The Flux Thimble Tube Inspection program does not prevent degradation due to aging effects but provides measures for inspection and evaluation to detect the degradation prior to loss of intended function. The program implements the recommendations of NRC Bulletin 88-09 (Reference 19).

All flux thimble tubes are currently inspected during each refueling outage. Wall thickness measurements are trended and wear rates are calculated. If the current measured wear exceeds the acceptance criteria or the predicted wear (as a measure of percent through wall), or if for a given flux thimble tube is projected to exceed the established acceptance criteria for wall thickness prior to the next refueling outage, corrective actions are taken to reposition, cap, or replace the tube. Program documentation maintains details regarding the core location, wear location, and the number of times a tube has been previously repositioned or replaced. Any thimble tube exhibiting an abnormally high wear rate is capped or replaced. Design changes are also implemented to use more wear-resistant thimble tube materials (e.g., chrome-plated stainless steel). The inspection frequency may be revised as appropriate based upon items such as operating experience and recommendations from the PWROG.

4.2.3 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. The limits imposed by the Primary Water Chemistry Plan (Reference 9) at DCPP Unit 2 are consistent with the EPRI Primary Water Chemistry Guidelines as described in Section 4.1.

The original DCPP Unit 2 support pins were fabricated from *INCONEL®* alloy X-750 that was hot rolled, solution treated or annealed, 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 SCC and likely to fail during the lifetime of a nuclear power plant. To address the susceptibility the support pins were replaced with cold-worked 316 stainless steel, with a design and stress distribution modified to be highly resistant to SCC.

Support pins were replaced at DCPP Unit 2 and detailed descriptions of the replacement are retained in the plant records (Reference 20).

4.2.4 Upflow Conversion Modification Project

Performing the upflow conversion modification significantly reduces the pressure differential across the baffle plates and therefore reduces and eliminates potential the jet flow and resulting fuel assembly damage. By performing the upflow conversion, the baffle jetting phenomena is permanently eliminated as a source of fuel cladding damage.

An added benefit of this modification is that it provides the opportunity, while performing modifications to the reactor internals, to address the upper head temperature by performing the upper head temperature reduction modification. This modification will result in the bulk of the upper head fluid reducing its operating temperature from the hot leg temperature (T_{hot}) to the cold leg temperature (T_{cold}).

The upflow conversion modification entails installing 24 mechanical plugs into the flow holes of the lower internals core barrel and machining 20 holes in the top former plate of the lower internals. This has the effect of directing the flow that would have gone through the holes and down the former plate region to the bottom of and through the core. The new flow direction will be along the downcomer region outside the core barrel and up the former plate region, exiting this region through the 20 new holes in the top former plate.

The upper head temperature reduction modification removes the 6 specimen plugs that are installed in 6 holes through the flange of the core barrel and machines 6 new holes in the upper internals upper support plate directly above the core barrel flange holes to provide a direct flow path from the core barrel cold leg inlet region to the upper head region.

The upflow conversion modification was performed at DCPP Unit 2, and detailed descriptions of the modification are retained in the plant records (Reference 21).

4.2.5 Reactor Vessel Internals Program

The PWR Vessel Internals Program is a new program that will be implemented prior to the period of extended operation. The program relies on implementation of the inspection and evaluation guidelines in MRP-227-A and MRP-228 to manage the aging effects of the reactor vessel internals components. This program is discussed in PG&E Letter DCL-14-103 (Reference 34) and is used to manage: (a) cracking, including stress corrosion cracking, primary water stress corrosion cracking, irradiation-assisted stress corrosion cracking, and cracking due to fatigue/cyclical loading; (b) loss of material induced by wear; (c) loss of fracture toughness due to either thermal aging embrittlement, irradiation embrittlement, or void swelling; (d) dimensional changes due to void swelling or distortion; and loss of preload due to thermal and irradiation-enhanced stress relaxation or irradiation-enhanced creep.

4.3 INDUSTRY PROGRAMS

4.3.1 WCAP-14577, Aging Management for Reactor Internals

The WOG (now PWROG) topical report WCAP-14577 (Reference 8) contains a technical evaluation of aging degradation mechanisms and aging effects for Westinghouse RVI components. The LRA was completed using the interim methodology in WCAP-14577 (Reference 8). The TLAA for the LRA (Reference 13) references WCAP-14577. The WOG sent the report to the NRC staff to demonstrate that WOG member plant owners that subscribed to the WCAP could adequately manage effects of aging on RVI during the period of extended operation, using approved aging management methodologies of the WCAP to develop plant-specific aging management programs.

The aging management review for the DCPP Unit 2 internals was completed in accordance with the requirements of WCAP-14577 (Reference 8).

4.3.2 MRP-227, Reactor Internals Inspection and Evaluation Guidelines

MRP-227 (Reference 3), 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.

4.3.2.1 MRP-227 RVI Component Categorizations

MRP-227 used a screening and ranking process to aid in the identification of required inspections for specific RVI components. MRP-227 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 United States 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 as "Primary" components, "Expansion" components, "Existing Programs" components, or "No Additional Measures" components, as 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 are not intended to supersede any ASME B&PV Code Section XI 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.

4.3.2.2 NEI 03-08 Guidance within MRP-227

The industry program requirements of MRP-227 are classified in accordance with the requirements of the NEI 03-08 (Reference 14) protocols. The MRP-227 guideline includes Mandatory, Needed, and Good Practice elements as follows:

- **Mandatory**

There is one Mandatory element:

> *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).*

DCPP Unit 2 Applicability: MRP-227, Revision 0, was officially issued by the industry in December 2008. Therefore, an AMP was required to be developed by December 2011. PG&E satisfied this requirement via issuance of Revision 0 of this WCAP in December 2011.

- **Needed**

There are five Needed elements:

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.*

   DCPP Unit 2 Applicability: MRP-227-A augmented inspections have been appropriately incorporated into the DCPP RVI AMP program procedure TS1.ID11 (Reference 43). The Program Plan for Inspection of Reactor Vessel Internals will coordinate with the existing DCPP Unit 2 Inservice Inspection Program (Reference 2) and will supplement that program with the augmented examinations for managing the potential aging effects of the reactor vessel internals. The applicable Westinghouse tables contained in MRP-227-A are Table 4-3 (Primary), Table 4-6 (Expansion), Table 4-9 (Existing), and Table 5-3 (Acceptance Criteria and Expansion Criteria Recommendations) 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*].

   DCPP Unit 2 Applicability: Inspection standards will be in accordance with the requirements of MRP-228 (Reference 11). These inspection standards will be used for augmented inspection at DCPP Unit 2 as applicable where required by MRP-227 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.*

   DCPP Unit 2 Applicability: PG&E 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.*

   DCPP Unit 2 Applicability: As discussed in subsection 4.3.3, PG&E 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.*

DCPP Unit 2 Applicability: PG&E 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.

4.3.2.3 GALL AMP Development Guidance

It should be noted that MRP-227-A, Appendix A (Reference 3) also includes a description of the attributes that make up an acceptable AMP. These attributes are similar to the previously discussed attributes of Revision 2 of the GALL Report and are consistent with the PG&E Aging Management Review process. Evaluation of the DCPP Unit 2 RVI AMP against GALL attribute elements is provided in Section 5 of this program plan.

As part of License Renewal, PG&E agreed to participate in industry activities associated with the development of the standard Industry Guidelines 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 DCPP Unit 2 RVI AMP.

4.3.2.4 MRP-227-A Applicability to Diablo Canyon Power Plant Unit 2

The applicability of MRP-227-A to DCPP 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.*

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

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

  DCPP Unit 2 Applicability: DCPP Unit 2 operates as a base-load unit.

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

  DCPP 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 considered. There have been no modifications to reactor internals components at DCPP Unit 2 since May 2007.

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

4.3.3 WCAP-17451, Reactor Internals Guide Tube Wear – Westinghouse Domestic Fleet Operational Projections

In February 2015, the PWROG submitted Westinghouse topical report WCAP-17451-P (Reference 28) to the NRC for information only. The report documents the results of a PWROG program of which the purpose was 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 wear standpoint, as well as to establish an initial inspection schedule based on the various guide tube designs for the utilities that participated in the program.

As a result, the industry recognizes the document as providing the acceptance criteria for inspections of CRGT assembly guide cards.

4.3.4 Ongoing Industry Programs

The U.S. nuclear industry, through both the EPRI/MRP and the PWROG, continues to sponsor activities related to RVI aging management. PG&E will maintain cognizance of industry activities related to PWR internals inspection and aging management and will address and implement industry guidance stemming from those activities, as appropriate under NEI 03-08 practices.

4.4 SUMMARY

It should be noted that the PG&E, MRP, and 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 and then determination of the proper inspection or mitigation program that provides reasonable assurance that the components will continue to perform their intended functions through the period of extended operation. The GALL-based approach was used at DCPP for the initial basis of the LRA that resulted in the NRC SER (Reference 1).

The approach used to develop the DCPP Unit 2 AMP is fully compliant with regulatory directives and approved documents. The additional evaluations and analyses completed by the MRP industry group have provided clarification to 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 PG&E position that use of the AMR produced by the LRA methodology, 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 DIABLO CANYON POWER PLANT UNIT 2 REACTOR INTERNALS AGING MANAGEMENT PROGRAM ATTRIBUTES

The DCPP 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 (cracking)
- Primary water stress corrosion cracking (cracking)
- Irradiation-assisted stress corrosion cracking (cracking)
- Wear (loss of material)
- Fatigue (cracking)
- Thermal aging embrittlement (reduction in fracture toughness)
- Irradiation embrittlement (reduction in fracture toughness)
- Void swelling and irradiation growth (distortion)
- Thermal and irradiation-enhanced stress relaxation or irradiation-enhanced creep (loss of preload or loss of mechanical closure integrity)

The attributes of the DCPP Unit 2 Reactor Internals AMP and compliance with NUREG-1801 (GALL Report) and Section XI.M16A, "PWR Vessel Internals" (Reference 6), as updated via LR-ISG-2011-04, 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.

PG&E fully utilized the GALL process contained in NUREG-1801 (Reference 6) in performing the aging management review of the reactor internals in the license renewal process. However, PG&E made several commitments for DCPP Unit 2 (see Reference 1), as discussed in Section 1. This Program Plan for the Inspection of Reactor Vessel Internals is to conform to LR-ISG-2011-04 prior to the period of extended operation at DCPP Unit 2. Additionally, PG&E committed to submitting responses to the applicable aging management program plant-specific action items, conditions and limitations identified in the NRC Safety Evaluation, Revision 1, on MRP-227 to the NRC by December 2015.

This AMP is consistent with the GALL process and includes consideration of the augmented inspections identified in MRP-227-A. The requirements of the commitment are hereby fulfilled. Specific details of the DCPP Unit 2 Reactor Internals AMP are summarized in the following subsections.

5.1 GALL REVISION 2 PROGRAM ELEMENT 1: SCOPE OF PROGRAM

GALL Report AMP Element Descriptions

> *"The scope of the program includes all RVI components based on the plant's applicable nuclear steam supply system. The scope of the program applies the methodology and guidance in MRP-227-A, which provides an 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 Babcock & Wilcox (B&W), Combustion Engineering (CE), and Westinghouse. The scope of components considered for inspection in MRP-227-A include core support structures, 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). In addition, ASME Code, Section XI includes inspection requirements for PWR removable core support structures in Table IWB-2500-1, Examination Category B-N-3, which are in addition to any inspections that are implemented in accordance with MRP-227-A.

The scope of the program does not include consumable items, such as fuel assemblies, reactivity control assemblies, and nuclear instrumentation. 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 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'." (Reference 6)

Diablo Canyon Power Plant Unit 2 Program Scope

The DCPP PWR Vessel Internals Program provides guidelines to adequately manage the aging effects of selected DCPP reactor vessels internals components, both non-bolted and bolted. The DCPP Unit 2 RVI consist of three basic assemblies: (1) the upper core support structure that is removed during each refueling operation to obtain access to the reactor core, (2) the lower core support structure that can be removed, if desired, following a complete core unload, and (3) the incore instrumentation support structures. Additional RVI details are provided in subsection 4.2.2 of the DCPP Updated Final Safety Analysis Report (UFSAR) (Reference 22).

The DCPP PWR Vessel Internals Program will be focused on managing age related degradation mechanisms by performing inspections intended to identify crack initiation and growth due to IASCC and cracking due to fatigue/cyclical loading, loss of material induced by wear, PWSCC and SCC; reduction of fatigue toughness due to IE, TE and void swelling; changes in dimensions due to void swelling; and loss of preload due to thermal and irradiation-enhanced stress relaxation or irradiation-enhanced creep, in reactor vessel internals components.

The DCPP PWR Vessel Internals Program applies the guidance in MRP-227-A, which identified and provides the basis for the required augmented inspections, inspection techniques to permit detection and characterization of aging effects of interest, prescribed frequency of inspections, and examination acceptance criteria for assuring the functional integrity of Westinghouse reactor vessel internals. The program scope includes the Westinghouse plants "Primary" components listed in Table 4-3 of MRP-227-A and Westinghouse plants "Expansion" components listed in Table 4-6 of MRP-227-A. The aging effects of a third set of reactor vessel internals locations, consistent with those listed in Table 4-9 of MRP-227-A, are adequately managed by the existing DCPP programs. Those reactor vessel internals components for which the effects of all aging mechanisms were determined by MRP-227-A to be below the screening criteria were placed in the "No Additional Measures" group. No additional aging management is necessary for the reactor vessel internals components in the No Additional Measures group. In no case does the No Additional Measures determination supersede the ASME Section XI Inservice Inspection requirements for components in this group.

The DCPP Unit 2 RVI subcomponents that required aging management review are indicated in the previously submitted Table 2.3.1-1 of the DCPP LRA (References 13 and 34). These components were subjected to an aging management review, and the results of this review were presented in Table 3.1.2-1 of the DCPP LRA. Specific columns of LRA Table 3.1.2-1 are reproduced in Appendix B as Table B-1, which includes all of the subcomponents of the RVI that required aging management review along with the related NUREG-1801 item(s) and the relevant Table 3.1.1 item from the LRA.

DCPP LRA Table 3.1.2-1 provides the detailed results of the reactor internals aging management review. The table identifies the aging effects that require management for those components requiring review. A column in the tables lists the programs and activities at DCPP Unit 2 that are credited to address the aging effects for each component 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 (Reference 1).

The results of the industry research provided by MRP-227-A, summarized in the tables in 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 DCPP Unit 2 RVI AMP scope is based on previously established and approved GALL Report approaches through application of the WCAP-14577 methodologies to determine those components that require aging management. Likewise, the additional information provided in the industry guidelines document, MRP-227-A (results of Appendix C), is rooted in the GALL methodology and provides a basis for augmented inspections that were required to complete this DCPP Unit 2 RVI AMP by providing the inspection method, frequency of inspection, and examination acceptance criteria.

Conclusion

This element complies with the corresponding aging management attribute in NUREG-1801, Section XI.M16A and License Renewal Commitments 72 and 73 (Reference 34).

5.2 GALL REVISION 2 PROGRAM ELEMENT 2: PREVENTIVE ACTIONS

GALL Report AMP Element Descriptions

> *"MRP-227-A 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, as described in GALL AMP XI.M2, 'Water Chemistry'."* (Reference 6)

Diablo Canyon Power Plant Unit 2 Preventive Actions

The DCPP PWR Vessel Internals Program does not prevent degradation due to aging effects; rather, it provides measures for monitoring to detect degradation prior to loss of intended function. Preventative measures to mitigate aging effects such as loss of material and cracking in the primary water system are established and implemented in accordance with the DCPP Water Chemistry Program (Reference 15).

Primary Water Chemistry Plan

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, etc.) 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 Diablo Canyon Power Plant Primary Strategic Water Chemistry Plan (Reference 15) is based on the current, approved revisions of EPRI PWR Primary Water Chemistry Guidelines (Reference 9).

This program is consistent with the corresponding program described in the GALL Report (References 6 and 20).

The limits of known detrimental contaminants imposed by the chemistry monitoring program are consistent with the EPRI PWR Primary Water Chemistry Guidelines (Reference 9).

Conclusion

This element complies with the corresponding aging management attribute in NUREG-1801, Section XI.M16A and License Renewal Commitments 72 and 73 (Reference 34).

5.3 GALL REVISION 2 PROGRAM ELEMENT 3: PARAMETERS MONITORED OR INSPECTED

GALL Report AMP Element Descriptions

> *"The program manages the following age-related degradation effects and mechanisms that are applicable in general to RVI components at the facility: (a) cracking induced by SCC, PWSCC, IASCC, or fatigue/cyclic 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 dimensions due to void swelling, or distortion; and (e) loss of preload due to thermal and irradiation-enhanced stress relaxation or creep.*
>
> *For the management of cracking, the program monitors the evidence of surface breaking linear discontinuities if a visual inspection technique is used as the non-destructive examination (NDE) method, or for relevant flaw presentation signals if a volumetric ultrasonic testing (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. Instead, the impact of loss of fracture toughness on component integrity is indirectly managed by: (1) using visual or volumetric examination techniques to monitor for cracking in the components and (2) applying applicable*

reduced fracture toughness properties in the flaw evaluations, in cases where cracking is detected in the components and is extensive enough to necessitate a supplemental flaw growth or flaw tolerance evaluation. The program uses physical measurements to monitor for any dimensional changes due to void swelling or distortion.

Specifically, the program implements the parameters monitored/inspected criteria consistent with the applicable tables in Section 4, 'Aging Management Requirements,' in MRP-227-A." (Reference 6)

Diablo Canyon Power Plant Unit 2 Parameters Monitored or Inspected

The DCPP PWR Vessel Internals program monitors the following aging effects by inspection, in accordance with the guidance of MRP-227-A or ASME Code Section XI, Category B-N-3:

1) Cracking

   Cracking is due to SCC, PWSCC, IASCC or fatigue/cyclical loading. Cracking is monitored with a visual inspection for evidence of surface-breaking linear discontinuities or a volumetric examination. Surface examinations may also be used to supplement visual examinations for detection and sizing of surface-breaking discontinuities.

2) Loss of Material

   Loss of material is due to wear. Loss of material is monitored with a visual inspection for gross or abnormal surface conditions.

3) Loss of Fracture Toughness

   Loss of fracture toughness is due to TE or IE. The impact of loss of fracture toughness on component integrity is indirectly managed by monitoring for cracking by using visual or volumetric examination techniques, and by applying applicable reduced fracture toughness properties in flaw evaluations if any detected cracking is determined to be extensive enough to necessitate a supplemental flaw growth or flaw tolerance evaluation.

4) Changes in Dimension

   Changes in dimension are due to void swelling or distortion. The program supplements visual inspection with physical measurements to monitor for any dimensional changes due to void swelling or distortion.

5) Loss of Preload

   Loss of preload is due to thermal and ISR or irradiation-enhanced creep. Loss of preload is monitored with a visual inspection for gross surface conditions that may be indicative of loosening in applicable bolted, fastened, keyed or pinned connections.

The DCPP PWR Vessel Internals program manages the aging effects noted above consistent with the guidance designated for the Westinghouse-designed Primary components included in Table 4-3 of MRP-227-A and the Westinghouse-designed Expansion components included in Table 4-6 of MRP-227-A.

For license renewal, the ASME Section XI Program consists of periodic volumetric, surface, and/or visual examination of components for assessment, signs of degradation, and corrective actions. This program is consistent with the corresponding program described in the GALL Report (Reference 6).

Appendices B and C of the DCPP Unit 2 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 and License Renewal Commitments 72 and 73 (Reference 34).

5.4 GALL REVISION 2 PROGRAM ELEMENT 4: DETECTION OF AGING EFFECTS

GALL Report AMP Element Descriptions

"The inspection methods are defined and established in Section 4 of MRP-227-A. Standards for implementing the inspection methods are defined and established in MRP-228. In all cases, well-established inspection methods are selected. These methods include volumetric UT examination methods for detecting flaws in bolting 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). VT-3 visual methods may be applied for the detection of cracking in non-redundant RVI components only when the flaw tolerance of the component, as evaluated for reduced fracture toughness properties, is known and the component has been shown to be tolerant of easily detected large flaws, even under reduced fracture toughness conditions. VT-3 visual methods are acceptable for the detection of cracking in redundant RVI components (e.g., redundant bolts or pins used to secure a fastened RVI assembly).

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.

The program adopts the guidance in MRP-227-A for defining the 'Expansion Criteria' that needed to be applied to the inspection findings of 'Primary' components and for expanding the examinations to include additional 'Expansion' components. RVI component inspections are *performed consistent with the inspection frequency and sampling bases for 'Primary' components, 'Existing Programs' components, and 'Expansion' components in MRP-227-A.*

In some cases (as defined in MRP-227-A), 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 dimensions due to void swelling or distortion.

Inspection coverages for 'Primary' and 'Expansion' RVI components are implemented consistent with Sections 3.3.1 and 3.3.2 of the NRC SE, Revision 1, on MRP-227." (Reference 6)

Diablo Canyon Power Plant Unit 2 Detection of Aging Effects

The DCPP PWR Vessel Internals program detects the aging effects listed in Element 3 through performance of examinations of the parameters specified in MRP-227-A, Table 4-3 for Westinghouse-designed Primary components and for parameters specified in MRP-227-A, Table 4-6 for Westinghouse-designed expansion components.

The DCPP PWR Vessel Internals program provides both examination acceptance criteria for conditions detected during inspection of Westinghouse-designed Primary components, as well as criteria that are applied to determine if scope expansion of examinations is required. When the examination acceptance criteria for the Westinghouse designed Primary components included in MRP-227-A, Table 4-3 are not met, the program requires expanding the scope of examinations to include the additional Westinghouse-designed Expansion components included in MRP-227-A, Table 4-6.

MRP-227-A included a fourth group of components designated as requiring No Additional Measures. The aging of these components was determined to be negligible relative to other reactor internals, and therefore, surface breaking discontinuities and cracking caused by SCC, IASCC and fatigue.

VT-3 examinations are applied to detection of cracking only when the flaw tolerance of the component or affected assembly, as evaluated for reduced fracture toughness, has been shown to be tolerant of easily detectable large flaws, even under reduced fracture toughness conditions. VT-3 examinations may also be used to inspect for loss of material that is induced by wear, and other aging effects such as gross distortion caused by void swelling and irradiation growth, and aging effects of loss of preload that is caused by thermal and irradiation-enhanced stress relaxation and creep.

Surface measurements may be used to supplement visual examinations required by this program to reject or accept relevant indications.

The impact of loss of fracture toughness (due to TE or IE) on component integrity is indirectly managed by monitoring for cracking using visual or volumetric examination techniques and by applying applicable reduced fracture toughness properties in the flaw evaluations after cracking is determined to be extensive enough to warrant a supplemental flaw growth or flaw tolerance evaluation.

One hundred percent of the accessible volume/area of each component will be examined for the Primary and Expansion components inspection category components. The minimum examination coverage for primary and expansion inspection categories is 75 percent of the component's total (accessible plus inaccessible) inspection area/volume be examined. When addressing a set of like components (e.g., bolting), the minimum examination coverage for primary and expansion inspection categories is 75 percent of the component's total population of like components (accessible plus inaccessible).

If conditions are detected during the examination, DCPP will enter the information into the corrective action program and evaluate whether the results of the examination ensure that the component (or set of components) will continue to meet the intended function under all licensing basis conditions of operation until the next scheduled examination. Engineering evaluations that demonstrate the acceptability of a detected condition will be performed consistent with WCAP-17096-NP.

Detection of indications required by the ASME Section XI ISI Program is well-established and field-proven through 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.

Inspection can be used to detect physical effects of degradation in both CASS and non-CASS components, 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 visual techniques include VT-3, and EVT-1 (enhanced visual test). 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 in augmented reactor internals inspections are documented in MRP-228.

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

In the augmented inspections detailed in the 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. This includes both CASS and non-CASS components. 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, "Reactor Internals Acceptance Criteria Methodology and Data Requirements" (Reference 12). 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 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 five 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 five 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.

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 the strategy proposed by MRP-227-A, UT inspections have been recommended exclusively for detection of flaws in bolts.

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 minimum acceptable bolting patterns. The evaluation program must demonstrate that the remaining bolts meet the requirements for a minimum 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 minimum acceptable bolting pattern before the next inspection.

Establishment of the minimum 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 inspection of bolts assumes that the plant owner will work with the designer to establish minimum acceptable bolting patterns prior to the inspection to support continued operation. For Westinghouse-designed plants, minimum acceptable bolting patterns for baffle-former and barrel-former bolts are available through the PWROG. PG&E has been a full participant in the development of the PWROG documents and has full access and use.

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. Direct physical measurements are required only for the hold down springs constructed of Type 304 stainless steel. However, the hold down spring for DCPP Unit 2 is constructed of Type 403 SS, which does not require inspection under MRP-227 (Reference 3). Therefore, no direct physical measurement is required for DCPP Unit 2.

Conclusion

This element complies with or exceeds the corresponding aging management attribute in NUREG-1801, Section XI.M16A and License Renewal Commitments 72 and 73 (Reference 34).

5.5 GALL REVISION 2 PROGRAM ELEMENT 5: MONITORING AND TRENDING

GALL Report AMP Element Descriptions

> *"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-A and its subsections. Flaw evaluation methods including recommendations for flaw depth sizing and for crack growth determinations as well as for performing applicable limit load, linear elastic and elastic-plastic fracture analyses of relevant flaw indications, are defined in MRP-227-A. The examination and re-examinations that are implemented in accordance with MRP-227-A guidance, together with the criteria specified in MRP-228 for inspection methodologies, inspection procedures, and inspection personnel, provide for timely detection, reporting, and implementation of corrective actions for the aging effects and mechanisms managed by the program.*

The program applies applicable fracture toughness properties, including reductions for thermal aging or neutron embrittlement, in the flaw evaluations of the components in cases where cracking is detected in a RVI component and is extensive enough to warrant a supplemental flaw growth or flaw tolerance evaluation.

For singly-represented components, the program includes criteria to evaluate the aging effects in the inaccessible portions of the components and the resulting impact on the intended function(s) of the components. For redundant components (such as redundant bolts, screws, pins, keys, or fasteners, some of which are accessible to inspection and some of which are not accessible to inspection), the program includes criteria to evaluate the aging effects in the population of components that are inaccessible to the applicable inspection technique and the resulting impact on the intended function(s) of the assembly containing the components." (Reference 6)

Diablo Canyon Power Plant Unit 2 Monitoring and Trending

The methods for monitoring, recording, evaluating and trending the date that result from the DCPP PWR Vessel Internals program's inspections are in accordance with the evaluation methodologies detailed in MRP-227-A, Section 6. This includes the recommended evaluation methodologies for flaw depth sizing and crack growth determinations, as well as for performing applicable limit load, linear elastic and elastic-plastic fracture analyses of relevant flaw indications.

The DCPP PWR Vessel Internals program applies applicable fracture toughness properties, including reductions for thermal aging or neutron embrittlement, in the flaw evaluations of the components in cases where cracking is detected in a reactor vessel internal component and is extensive enough to warrant a supplemental flaw growth or flaw tolerance evaluation.

In accordance with MRP-227-A, the DCPP PWR Vessel Internals program includes criteria to evaluate the aging effects in the inaccessible portions of the components and the resulting impact on the intended function(s) of the components. For redundant components, the program includes criteria to evaluate the aging effects in the population of components that are inaccessible to the applicable inspection technique and the resulting impact on the intended function(s) of the assembly containing the components.

Examination and re-examinations are implemented in accordance with MRP-227-A, together with the criteria 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 age related degradation mechanisms within the scope of the program.

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 operating experience 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 are used to identify trends in materials degradation and forecast potential component degradation. The industry continues to share both material test data and operating experience through the auspices of the MRP and PWROG. PG&E has in the past and will continue to maintain cognizance of industry activities and will continue to share operating experience information related to PWR internals inspection and aging management.

Inspections credited in Appendix B are based on utilizing both the DCPP Unit 2 10-year ISI program and the augmented inspections derived from the industry program documented in MRP-227-A (Reference 3) and contained in Appendix C for reference purposes. 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.

Reporting requirements are included as part of the MRP-227-A guidelines (see subsection 4.3.2.2 of this AMP). 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 and License Renewal Commitments 72 and 73 (Reference 34).

5.6 GALL REVISION 2 PROGRAM ELEMENT 6: ACCEPTANCE CRITERIA

GALL Report AMP Element Descriptions

> *"Section 5 of MRP-227-A, which includes Table 5-1 for B&W-designed RVIs, Table 5-2 for CE-designed RVIs, and Table 5-3 for Westinghouse-designed RVIs, provides the specific examination and flaw evaluation acceptance criteria for the 'Primary' and 'Expansion' RVI component examination methods. For RVI components addressed by examinations performed in accordance with the ASME Code, Section XI, the acceptance criteria in IWB-3500 are applicable. For RVI components covered by other 'Existing Programs,' the acceptance criteria are described within the applicable reference document.*
>
> *As applicable, the program establishes acceptance criteria for any physical measurement monitoring methods that are credited for aging management of particular RVI components."* (Reference 6)

Diablo Canyon Power Plant Unit 2 Acceptance Criteria

The DCPP PWR Vessel Internals program acceptance criteria for the Westinghouse-designed Primary and Expansion component examinations are consistent with MRP-227-A, Section 5A. For the Westinghouse-designed Expansion components, ASME Code, Section XI, Section IWB-3500 acceptance criteria apply. The DCPP PWR Vessel Internals program establishes acceptance criteria for any physical measurement monitoring methods credited for aging management of particular reactor vessel internals components.

Those recordable indications that are the result of inspections required by the existing DCPP Unit 2 ISI program scope are evaluated in accordance with the applicable requirements of the ASME Code through the existing Corrective Action Program (Reference 23).

Inspection acceptance and expansion criteria are provided in 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 DCPP Unit 2 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, 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 a minimum bolting approach 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 (Reference 23), 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. Additional analysis to establish Appendix C expansion component evaluation criteria is being performed through the efforts of the PWROG. Status is monitored through direct PG&E 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 and XI.M16A and License Renewal Commitments 72 and 73 (Reference 34).

5.7 GALL REVISION 2 PROGRAM ELEMENT 7: CORRECTIVE ACTIONS

GALL Report AMP Element Descriptions

> *"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. The implementation of the guidance in MRP-227-A, 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 actions bases may be used to disposition relevant conditions if they have been previously approved or endorsed by the NRC. Alternative corrective actions not*

approved or endorsed by the NRC will be submitted for NRC approval prior to their implementation." (Reference 6)

Diablo Canyon Power Plant Unit 2 Corrective Action

Any detected conditions that do not satisfy the examination acceptance criteria are required to be dispositioned through the DCPP corrective action program. The disposition will ensure that licensing and design basis functions of the reactor internals will be continued to be fulfilled.

The following corrective actions are suggested for the disposition of detected conditions that exceed the examination acceptance criteria:

1) Supplemental examinations to further characterize and potentially dispose of a detected condition

2) Engineering evaluation that demonstrates the acceptability of a detected condition

3) Repair, in order to restore a component with a detected condition to acceptance status

4) Replacement of a component with an unacceptable detected condition

If an engineering evaluation is used to disposition an examination result that does not meet the examination acceptance criteria, the engineering evaluation shall be conducted per an NRC-approved evaluation methodology.

DCPP QA procedures and review and approval processes are implemented in accordance with the requirements of 10 CFR 50, Appendix B and include administrative controls, as described in DCPP FSAR, Section 17.2, and provisions that specify when follow-up actions are required to be taken to verify that corrective actions are effective and those implemented to address significant conditions adverse to quality are effective in preventing recurrence of the condition.

The existing DCPP Unit 2 procedures for corrective actions (References 23, 24, and 25) and Inservice Repair and Replacement (Reference 26) will be credited for this element. The procedure in Reference 26 establishes the DCPP Unit 2 repair and replacement requirements for ASME Code Section XI, "Rules for Inservice Inspection of Nuclear Power Plant Components" (Reference 5). These requirements include the identification of a repair cycle, repair plan, and verification of acceptability for replacements. The corrective actions for augmented inspections at DCPP Unit 2 will be developed using repair and replacement procedures equivalent to those requirements in ASME B&PV Code, Section XI.

Conclusion

This element complies with the corresponding aging management attribute in NUREG-1801, Section XI.M16A and License Renewal Commitments 72 and 73 (Reference 34).

5.8 GALL REVISION 2 PROGRAM ELEMENT 8: CONFIRMATION PROCESS

GALL Report AMP Element Descriptions

> *"Site quality assurance procedures, review and approval processes, and administrative controls are implemented in accordance with the recommendations of NEI 03-08 and the requirements of 10 CFR Part 50, Appendix B, or their equivalent, as applicable. The implementation of the guidance in MRP-227-A, in conjunction with NEI 03-08 and other guidance documents, reports, or methodologies referenced in this AMP, provides and acceptable level of quality and an acceptance basis for confirming the quality of inspections, flaw evaluations, and corrective actions."* (Reference 6)

Diablo Canyon Power Plant Unit 2 Confirmation Process

DCPP QA procedures and review and approval processes are implemented in accordance with the requirements of 10 CFR 50 Appendix B and include administrative controls, as described in DCPP FSAR, Section 17.2, and provisions that specify when follow-up actions are required to be taken to verify that corrective actions are effective and those implemented to address significant conditions adverse to quality are effective in preventing recurrence of the condition.

The implementation of the guidance in MRP-227-A, in conjunction with the requirements of NEI 03-08 and other guidance documents, reports or methodologies referenced in this AMP, provides an acceptable level of quality and an acceptable basis for confirming the quality of inspection, flaw evaluation and other elements of aging management of the DCPP PWR Vessel Internals.

DCPP Unit 2 has an established 10 CFR Part 50, Appendix B, Program (Reference 27) that addresses the elements of corrective actions, confirmation process, and administrative controls. The DCPP 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. Recommendations from NEI 03-08 are considered in developing procedures and processes.

Conclusion

This element complies with or exceeds the corresponding aging management attribute in NUREG-1801, Section XI.M16A and License Renewal Commitments 72 and 73 (Reference 34).

5.9 GALL REVISION 2 PROGRAM ELEMENT 9: ADMINISTRATIVE CONTROLS

GALL Report AMP Element Descriptions

"The administrative controls for these types of programs, including their implementing procedures and review and approval processes, are implemented in accordance with the recommended industry guidelines and criteria in NEI 03-08, and are under existing site 10 CFR 50 Appendix B, Quality Assurance Programs, or their equivalent, as applicable. The evaluation in Section 3.5 of the NRC's SE,

Revision 1, on MRP-227-A provides the basis for endorsing NEI 03-08. This includes endorsement of the criteria in NEI 03-08 for notifying the NRC of any deviation from the I&E methodology in MRP-227-A and justifying the deviation no later than 45 days after its approval by a licensee executive." (Reference 6)

Diablo Canyon Power Plant Unit 2 Administrative Controls

See the evaluation in Section 5.8.

Conclusion

This element complies with or exceeds the corresponding aging management attribute in NUREG-1801, Section XI.M16A and License Renewal Commitments 72 and 73 (References 34).

5.10 GALL REVISION 2 PROGRAM ELEMENT 10: OPERATING EXPERIENCE

GALL Report AMP Element Descriptions

> *"The review and assessment of relevant operating experience for impacts on the program, including implementing procedures, are governed by NEI 03-08 and Appendix A of MRP-227-A. Consistent with MRP-227-A, the reporting of inspection results and operating experience treated as a "Needed" category item under the implementation of NEI 03-08.*
>
> *The program is informed and enhanced when necessary through the systematic and ongoing review of both plant-specific and industry operating experience, as discussed in Appendix B of the GALL report, which is documented in LR-ISG-2011-05."* (Reference 6)

Diablo Canyon Power Plant Unit 2 Operating Experience

Extensive industry and DCPP Unit 2 operating experience 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" (Reference 29) and 98-11, "Cracking of Reactor Vessel Internal Baffle Former Bolts in Foreign Plants" (Reference 30). 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 operating experience 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 are expected to discover general internals structure degradation. To date, very little degradation has been observed industry wide.

The systematic and ongoing review and assessment of relevant DCPP-specific and industry operating experience for its impact to the program are governed by NEI 03-08, "Guideline for the Management of Materials Issues" and MRP-227-A, Appendix A.

Based on industry operating experience, DCPP Unit 2 proactively replaced the originally installed Alloy X-750 guide tube support pins (split pins) with strain hardened (cold worked) 316 stainless steel pins in 2006 to reduce the susceptibility for SCC in the support pins.

DCPP Unit 2 is a converted upflow plant. Performing the upflow conversion modification significantly reduces the pressure differential across the baffle plates, and, therefore, reduces and eliminates potential jet flow and resulting fuel assembly damage. By performing the upflow conversion, the baffle jetting phenomena is permanently eliminated as a source of fuel cladding damage.

Relatively few incidents of PWR internals aging degradation have been reported in operating U.S. commercial PWR plants, and a summary of observations is maintained in Appendix A of MRP-227-A. With expectation of the ASME Section Inservice Inspection portions, the DCPP PWR Vessel Internals program is a new program. A key element of the program defined in MRP-227-A is the requirement for utilities to continue to report aging effects of PWR vessel internal components identified during examination. DCPP, through its participation in PWR Owners Group and EPRI-MRP activities, will continue to benefit from the reporting of examination information and results, and will share its own operating experience with the industry through those groups.

Industry operating experience is routinely reviewed by PG&E engineers using Institute of Nuclear Power Operations (INPO) Operating Experience (OE), the Nuclear Network, and other information sources, as directed under the applicable procedure (References 31 and 32), 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. PG&E will continue to 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.

Conclusion

This element complies with or exceeds the corresponding aging management attribute in NUREG-1801, Section XI.M16A and License Renewal Commitments 72 and 73 (Reference 34).

6 DEMONSTRATION

PG&E has demonstrated a long-term commitment to aging management of reactor internals at DCPP Unit 2. 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 DCPP Unit 2 reactor vessel internals have been performed during each 10-year interval since plant operations commenced.
- As documented in PG&E administrative procedures, reports are continuously reviewed by DCPP Unit 2 personnel with PG&E for applicable issues that indicate operating procedures or programs require updates based on new OE.
- Review of Quality Verification Department audit reports, NRC inspection reports, and INPO evaluations indicate no unacceptable issues related to reactor vessel internals inspections.
- The Primary Water Chemistry Program (Reference 15) at DCPP Unit 2 has been effective in maintaining oxygen, halogens, and sulfate at levels sufficiently low to prevent SCC of the reactor vessel internals.
- The control rod guide tube support pins (Reference 20) at DCPP Unit 2 were replaced.
- An upflow conversion (Reference 21) was completed at DCPP Unit 2.
- PG&E has actively participated in past and ongoing EPRI and PWROG RVI activities. PG&E will continue to 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.

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 degradation at its first appearance, consistent with the ASME 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.

DCPP Unit 2 will enter the period of extended operation at midnight on August 26, 2025 in conjunction with Cycle 24. In compliance with MRP-227-A requirements, the augmented inspections discussed in this AMP have been incorporated into the DCPP RVI AMP program procedure (Reference 43), along with other applicable DCPP procedures used to perform the ASME Section XI examinations, and implemented according to the requirements of MRP-227-A and Appendix C of this AMP. As discussed, the industry MRP-227-A guidelines also provide updates as experience is gained through inspection results. This feedback loop will enable updates based on actual inspection experience.

The DCPP Unit 2 Inspection Program Plan for Reactor Vessel Internals is comprised of the augmented inspections described in this document, as summarized in Appendix C, combined with the ASME Section XI ISI program inspections. In addition to existing DCPP Unit 2 programs and use of Operating Experience Reports (OERs), provide reasonable assurance that the reactor internals will continue to perform their intended functions through the period of extended operation.

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

Table 6-1 lists the compliance of topical report conditions to the SE on MRP-227.

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 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 DCPP Unit 2
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	In Table C-1 for the baffle-former bolts, the inspection frequency was changed from 10 to 15 additional effective full-power years (EFPY) to subsequent examination on a ten-year interval.
6. Periodicity of the re-examination of "Expansion" inspection category components	Applicable	"Re-inspection every 10 years following initial inspection" was added to every component under the Examination Method/Frequency column in Table C-2.
7. Updating of MRP-227, Revision 0, Appendix A	Applicable	Section 5 is updated to reflect XI.M16A from GALL Revision 2 (Reference 6).

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." (Reference 3)

DCPP Unit 2 Compliance

The process used to verify that DCPP 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:

1. Identify typical Westinghouse PWR internals components (MRP-191, Table 4-4).
2. Identify DCPP Unit 2 PWR internals components.
3. Compare typical Westinghouse PWR internals components to the DCPP Unit 2 PWR internals components.
    a. Confirm that no additional items were identified by this comparison (primarily supports A/LAI 2).
    b. Confirm that the materials identified for DCPP Unit 2 are consistent with those materials identified in MRP-191, Table 4-4.
    c. Confirm that the DCPP Unit 2 internals are the same as, or equivalent to, the typical Westinghouse PWR internals regarding design and fabrication.
4. Confirm that the DCPP Unit 2 operating history is consistent with the assumptions in MRP-227-A regarding core loading patterns.
5. Confirm that the DCPP Unit 2 RVI materials operated at temperatures within the original design basis parameters.
6. Determine stress values based on design basis documents.
7. Confirm that any changes to the DCPP Unit 2 RVI components do not impact the application of the MRP-227-A generic aging management strategy.

DCPP 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. DCPP Unit 2 operating history is consistent with the assumptions in MRP-227-A with regard to neutron fluence.

   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. The DCPP Unit 2 fuel management program changed from a high-leakage to a low-leakage core loading pattern prior to 30 years of operation. The first two cycles for DCPP Unit 2 were traditional core, with some feed assemblies on the periphery. Starting in cycle 3+, the core design changed to a low-leakage loading pattern. By operating with a high-leakage core design for less than 30 years of operation, DCPP Unit 2 meets the fluence and fuel management assumptions in MRP-191 and requirements for MRP-227-A application.

   b. DCPP Unit 2 has operated under base-load conditions over the life of the plant. Therefore, the actual number of unit loading and unloading occurrences is expected to be a small fraction of the cycles assumed in the fatigue analyses. Due to the infrequent nature of this cyclic transient, and due to the large margin to the assumed number of occurrences, it is not necessary to track the occurrence (Reference 35). Since DCPP Unit 2 operates at base load, assumptions in MRP documents regarding operational parameters affecting fluence are satisfied.

2. The DCPP Unit 2 RCS operates between T_{hot} and T_{cold}, which are not lower than 531.9°F for T_{cold} and not higher than 610.1°F for T_{hot} (Reference 22). The design temperature for the reactor vessel is 650°F (Reference 22). DCPP Unit 2 operating history is within original design basis parameters; therefore, it is consistent with the assumptions used to develop the MRP-227-A aging management strategy with regard to temperature operational parameters.

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

   a. No additional components are identified that adversely affect the MRP-191 FMECA process for DCPP Unit 2 and the components required to be in the DCPP Unit 2 RVI program (Reference 13) are consistent with those contained in MRP-191 (Reference 10).

   b. Most of the materials for DCPP Unit 2 are identical to, or equivalent with, those materials identified in MRP-191, Table 4-4 for Westinghouse-designed plants (Reference 10). The exceptions are the upper instrumentation brackets, clamps, terminal blocks, and conduit straps (conduit support, conduit support gusset, gusset clamp, and thermocouple stop), which are identified as having a material different that that specified in MRP-191 and involve CF8. An expert panel was conducted to disposition these component material differences (Reference 36). Several additional components have different materials than those specified in MRP-191; however, they have been determined to have no effect of the recommended MRP aging management inspection sampling strategy.

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

4. Modifications to the DCPP Unit 2 reactor internals include a flux thimble tube inspection and replacement program (Reference 18), a control rod guide tube support pin replacement project (Reference 20), and an upflow conversion modification project. Details of these replacements are retained in plant records. No additional modifications were made to the plant after 2011. The design has been maintained over the lifetime of the plant as specified by the original equipment manufacturer, operational parameters are compliant with MRP-227-A requirements with regard to fluence and temperature, and the components and materials are the same as those considered in MRP-191. Therefore, the DCPP 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 DCPP Unit 2 evaluation for A/LAI 1 of the NRC SE on MRP-227, Revision 0, confirms that MRP-227-A is applicable to DCPP Unit 2. Therefore, the requirements for application of MRP-227-A as a strategy for managing age-related material degradation in reactor internals components have been met.

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." (Reference 3)

DCPP Unit 2 Compliance

This A/LAI requires comparison of the RVI components that are within the scope of license renewal for DCPP Unit 2 to those components contained in MRP-191, Table 4-4. DCPP Unit 2 RVI components were tabulated (References 13 and 34) and compared to the typical Westinghouse PWR components in MRP-191, Table 4-4. From the review, it was determined that the components required to be in the DCPP Unit 2 program (Reference 13) are consistent with those contained in MRP-191.

Several components have different materials than those specified in the MRP-191 assessment. The potential for alternate materials, specifically CF8, to be used for the upper internals instrumentation brackets, clamps, terminal blocks and conduit straps was identified. An expert panel was conducted to disposition these component material differences (Reference 36). It was concluded that the material differences result in No Additional Measures or changes to the existing aging management program and MRP-227-A inspection schedule.

The completion of the expert panel supports the requirement that the AMP shall provide assurance that the effects of aging on the DCPP 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. Several additional components have slightly different materials specifications (i.e., different grades of austenitic stainless steel) than those specified in MRP-191; however, they have been determined to have no effect on the recommended MRP aging management inspection strategy.

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, are applicable to DCPP Unit 2 with no modifications.

Conclusion

DCPP Unit 2 complies with A/LAI 2 of the NRC SE on MRP-227, Revision 0, for all components as a result of completion of the expert panel.

Therefore, DCPP Unit 2 meets the requirement for application of MRP-227-A as a strategy for managing age-related material 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." (Reference 3)

DCPP Unit 2 Compliance

DCPP Unit 2 is compliant with the requirements in MRP-227-A, Table 4-9, as shown in Table C-3 of this document. This is detailed in the plant-specific DCPP program documents for ASME Section XI (Reference 2) and the plant-specific flux thimble program (Reference 18).

Conclusion

DCPP 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." (Reference 3)

DCPP Unit 2 Compliance

This Applicant/Licensee Action Item is not applicable to DCPP Unit 2 since it 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 DCPP 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." (Reference 3)

DCPP Unit 2 Compliance

DCPP utilizes a Type 403 SS hold down spring; therefore, PG&E is not required to perform physical measurements on the DCPP Unit 2 hold down spring. No other RVI components require a physical measurement according to MRP-227-A.

Conclusion

Applicant/Licensee Action Item 5 of the NRC SE on MRP-227, Revision 0, is not applicable to DCPP Unit 2.

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." (Reference 3)

DCPP Unit 2 Compliance

This Applicant/Licensee Action Item is not applicable to DCPP Unit 2 since it only applies to B&W plants.

Conclusion

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." (Reference 3)

DCPP Unit 2 Compliance

The NRC final SE on MRP-227 (Reference 3, subsection 3.3.7) states that, for assessment of CASS materials, the applicant/licensee for license renewal may apply the criteria in the NRC letter of May 19, 2000, License Renewal Issue No. 98-0030, "Thermal Aging Embrittlement of Cast Austenitic Stainless Steel Components" (Reference 38) 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 components' 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 engineering drawings for the DCPP Unit 2 control rod guide tube intermediate flanges permitted alternate material ASTM International Standard A351, Grade CF8. The DCPP Unit 2 standalone mixing devices, upper instrumentation conduit and supports – stops, supports, gussets, and clamps; upper support column assemblies – bases (mixer and orifice base); lower support column bodies – caps; and BMI column assemblies – cruciform (standard and special) are CASS.

For completeness, it is noted that the DCPP Unit 2 lower support is plate (wrought) material.

For each of the CASS components, the elemental percentages from the chemical data retrieved from certified material test reports (CMTRs) for the CASS component are input into Hull's formula (per guidance of NUREG/CR-4513 [Reference 39]) to calculate the delta ferrite content of the CASS material. The CMTRs do not list the element percentage for nitrogen; thus, per the guidance of NUREG/CR-4513, nitrogen is assumed to be 0.04 percent (Reference 39). The CMTRs do not typically list an elemental percentage for molybdenum. CASS materials A351 and A296, Grade CF8 did not have a requirement for percent molybdenum at time of fabrication in 1968 (Reference 40). The 2013 Edition of the ASME Boiler & Pressure Vessel Code (Reference 41) has SA-351, Grade CF8 chemistry requirements that specify a maximum of 0.5 percent molybdenum; thus, this maximum value is conservatively input into Hull's formula. When CMTRs were not located, a conservative combination of SA-351, Grade CF8 chemical requirements was used to show that the ferrite content can potentially exceed 20 percent. The results of the TE evaluation for the DCPP Unit 2 CASS components are summarized in Table 6-2.

Based on the criteria of the NRC letter dated May 19, 2000 (Reference 38):

- The control rod guide tube intermediate flanges are considered to be potentially susceptible to TE.
- The upper instrumentation conduit and supports (stops, supports, gussets, and clamps) are considered to be potentially susceptible to TE.
- Twenty-eight of the 31 mixing devices are not susceptible to TE; three of the 31 are potentially susceptible to TE.

- The upper support column – bases (mixing style) are not susceptible to TE.
- The upper support column – bases (orifice style) are not susceptible to TE.
- Twenty-four of the twenty-six BMI column cruciforms are not susceptible to TE; two of the twenty-six are considered to be potentially susceptible to TE.
- Ninety-five of the ninety-six lower support column bodies – caps are not susceptible to TE; one of the ninety-six is considered to be potentially susceptible to TE.

All of the preceding components were considered in MRP-191 (Reference 10) and were screened for susceptibility to material degradation, including consideration of TE and IE. With the exception of the control rod guide tube intermediate flanges and the upper instrumentation conduit and supports (stops, supports, gussets, and clamps), the aforementioned components were screened as CASS and were considered for TE in MRP-191. As discussed in the response to A/LAI 2, the assessments of the CASS upper instrumentation conduit and supports (stops, supports, gussets, and clamps) were evaluated by an expert panel, taking into consideration their potential susceptibility to TE and their impact on the DCPP Unit 2 aging management strategy. The expert panel concluded that the material differences result in No Additional Measures or changes to the existing aging management program and MRP-227-A inspection schedule.

The DCPP Unit 2 hold down spring is 403 SS, a martensitic stainless steel. No martensitic precipitation-hardened stainless steel components were identified for the DCPP Unit 2 reactor vessel internals.

Table 6-2 Summary of Diablo Canyon Unit 2 CASS Components and Their Susceptibility to TE

CASS Component MRP-191 [10] Name	Material	Molybdenum Content (Percent)	Casting Method	Ferrite Content (Percent)	Susceptibility to TE (Based on NRC Letter [38])
Upper Internals Assembly					
Control Rod Guide Tube Assemblies and Flow Downcomers – Intermediate flanges	ASTM A240 or Alternate ASTM A351, Grade CF8	0.5 Maximum	Static(4)	Possible > 20%(2)	Potentially Susceptible(2)
Mixing Devices (Stand-alone)	ASTM A351, Grade CF8	0.5 Maximum	Static(4)	28 of 31 ≤ 20%(1)	28 of 31 Not Susceptible(1)
				3 of 31 Possible > 20%(2)	3 of 31 Potentially Susceptible(2)
Upper Support Column – Upper Instrumentation Conduit and Supports (Thermocouple stops, supports, gussets, and clamps(5))	ASTM A351, Grade CF8	0.5 Maximum	Static(4)	Possible > 20%(2)	Potentially Susceptible(2)
Upper Support Column Assemblies, Mixer Bases	ASTM A351, Grade CF8	0.5 Maximum	Static(4)	≤ 20%(1)	Not Susceptible(1)
Upper Support Column Assemblies, Column Bases	ASTM A351, Grade CF8	0.5 Maximum	Static(4)	12 of 13 ≤ 20%(1)	12 of 13 Not Susceptible(1)
				1 of 13 Possible > 20%(2)	1 of 13 Potentially Susceptible(2)
Lower Internals Assembly					
BMI Column Assemblies, Column Cruciform	ASTM A351, Grade CF8	0.5 Maximum	Static(4)	24 of 26 ≤ 20%(1)	24 of 26 Not Susceptible(1)
				2 of 26 Possible > 20%(2)	2 of 26 Potentially Susceptible(2)
Lower Support Column Bodies – Caps	ASTM A296 or ASTM A351, Grade CF8	0.5 Maximum	Static(4)	95 of 96 ≤ 20%(1)	95 of 96 Not Susceptible(1)
				One is Possible > 20%(2)(3)	1 is Potentially Susceptible(2)

Notes:

(1) Conclusion is based on CMTR chemistry data with molybdenum = 0.5 percent (based on ASTM International chemistry requirements) and nitrogen = 0.04 percent (per guidance of [39]) input into Hull's formula.

(2) When a CMTR was not located, a conservative combination of ASME SA-351, Grade CF8 chemical requirements input into Hull's formula [39] shows that the ferrite content can exceed 20 percent.

(3) A letter in the historical records provides certification that the original actual test result data were not available due to loss of specific traceability, because of the administrative systems used at the time of manufacture. Thus, actual chemistry data are not available for input into Hull's formula and the cast part is conservatively assessed per note (2).

(4) The part is assumed to have been static cast.

(5) The Diablo Canyon Unit 2 CASS parts on the upper instrumentation conduit and supports are the thermocouple stops, supports, gussets, and clamps.

Conclusion

It is concluded that continued application of the MRP-227-A (Reference 3) strategy will meet the requirement for managing age-related degradation of the Diablo Canyon Unit 2 CASS reactor vessel internals components.

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." (Reference 3).

DCPP Unit 2 Compliance

MRP-227-A identifies the following information that license renewal applicants must submit to the NRC for review and approval:

1. *An AMP for the facility that addresses the 10 program elements as defined in NUREG-1801, Revision 2, AMP XI.M16A.*

   DCPP Unit 2: The 10 programs elements are addressed via issuance of this AMP.

2. *To ensure the MRP-227 program and the plant-specific action items will be carried out by applicants/licensees, applicant/licensees are to submit an inspection plan which addresses the identified plant-specific action items for staff review and approval consistent with the licensing basis for the plant. If an applicant/licensee plans to implement an AMP which deviates from the guidance provided in MRP-227, as approved by the NRC, the applicant/licensee shall identify where the program deviates from the recommendations in MRP-227, as approved by the NRC, and shall provide a justification for any deviation which includes a consideration of how the deviation affects both "Primary" and "Expansion" inspection category components.*

   DCPP Unit 2: The inspection plan for DCPP Unit 2 is being submitted via this AMP.

3. *The regulation at 10 CFR 54.21(d) requires that an FSAR supplement for the facility contain a summary description of the programs and activities for managing the effects of aging and the evaluation of TLAAs for the period of extended operation. Those applicants for LR referencing MRP-227, as approved by the NRC, for their RVI component AMP shall ensure that the programs and activities specified as necessary in MRP-227, as approved by the NRC, are summarily described in the FSAR supplement.*

   DCPP Unit 2: Per the LRA (References 1 and 34), PG&E will maintain a summary list in the DCPP FSAR Update of activities that are required to manage the effects of aging for the systems, structures and components in the scope of the license renewal during the period of extended operation. The FSAR update will comply with 10 CFR 54.21(d).

4. *The regulation at 10 CFR 54.22 requires each applicant for LR to submit any technical specifications changes (and the justification for the changes) that are necessary to manage the effects of aging during the period of extended operation as part of its LRA. For the plant CLBs that include mandated inspection or analysis requirements for RVI either in the operating license for the facility or in the facility TS, the applicant/licensee shall compare the mandated requirements with the recommendations in the NRC-approved version of MRP-227. If the mandated requirements differ from the recommended criteria in MRP-227, as approved by the NRC, the conditions in the applicable license conditions or TS requirements take precedence over the MRP recommendations and shall be complied with.*

   DCPP Unit 2: Per the LRA (References 1 and 34), no technical specifications changes have been identified in order to manage the effects of aging of RVI during the period of extended operation.

5. *Pursuant to 10 CFR 54.21(c)(1), the applicant is required to identify all analyses in the CLB for their RVI components that conform to the definition of a TLAA in 10 CFR 54.3 and shall identify these analyses as TLAAs for the application in accordance with the TLAA identification requirement in 10 CFR 54.21(c)(1). MRP-227 does not specifically address the resolution of TLAAs that may apply to applicant/licensee RVI components. Hence, applicants/licensees who implement MRP-227, as approved by the NRC, shall still evaluate the CLB for their facilities to determine if they have plant-specific TLAAs that shall be addressed. If so, the applicant's/licensee's TLAA shall be submitted for NRC review along with the applicant/licensee's application to implement the NRC-approved version of MRP-227.*

   For those cumulative usage factor (CUF) analyses that are TLAAs, the applicant may use the PWR Vessel Internals Program as the basis for accepting these CUF analyses in accordance with 10 CFR 54.21(c)(1)(iii) only if the RVI components in the CUF analyses are periodically inspected for fatigue-induced cracking in the components during the period of extended operation. The periodicity of the inspections of these components shall be justified to be adequate to resolve the TLAA. Otherwise, acceptance of these TLAAs shall be done in accordance with either 10 CFR 54.21(c)(1)(i) or (ii), or in accordance with 10 CFR 54.21(c)(1)(iii) using the applicant's program that corresponds to NUREG-1801, Revision 2, AMP X.M1, "Metal Fatigue of Reactor Coolant Pressure Boundary Program." To satisfy the evaluation requirements of ASME Code, Section III, Subsection NG-2160 and NG-3121, the existing fatigue CUF analyses shall include the effects of the reactor coolant system water environment.

   DCPP Unit 2: Per the LRA (References 1 and 34) there are no analyses in the CLB for RVI that are TLAAs, in accordance with 10 CFR 54.3(a) Criteria 2 and 3.

DCPP Unit 2, per the Regulatory Issue Summary (Reference 37), is considered a Category D plant that is expected to submit their RVI AMP based on the guidance of MRP-227-A, which is consistent with their commitments. Per License Renewal commitment (Reference 34), DCPP Unit 2 must submit responses to MRP-227-A action items to the NRC by December 2015. DCPP Unit 2 must implement the PWR Reactor Internals Program to conform to LR-ISG-2011-04 prior to the period of extended operation.

7 PROGRAM ENHANCEMENT AND IMPLEMENTATION SCHEDULE

The requirements of MRP-227 are based on an 18-month refueling cycle and consider both effective full-power years (EFPY) and cumulative operation. The information contained in Table 7-1 is based on this information and includes a description of the currently projected scope of inspection pertaining to the reactor internals AMP. Should a change occur in plant operational practices or operating experience result in changes to the projections, appropriate updates will be performed on affected plant documentation in accordance with approved procedures. The results contained in Table 7-1 are based on average 20-month cycles in calculating the EFPY.

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

RFO	Cycle End Quarter/Year	Estimated EFPY	AMP-Related Scope	Inspection Method and Criteria	Comments
2R17	Spring / 2013	23.45			
2R18	Fall / 2014	24.92			
2R19	Spring / 2016	26.40	ASME Code Section XI ISI Initial MRP-227 augmented inspection for control rod guide tube guide cards(2)	ASME Code Section XI WCAP-17451-P is applied for control rod guide card inspections.	The initial inspection window for these components is no later than two refueling outages from the beginning of extended operation. DCPP Unit 2 has the option to perform these inspections until RO-26.
2R20	Spring / 2018	28.04			
2R21	Fall / 2019	29.58			
2R22	Spring / 2021	31.08			
2R23	Spring / 2023	32.72			
2R24	Fall / 2024	34.26	Initial MRP-227 augmented inspection for baffle-former bolts completed during or before this outage. (1)		Extended operation begins 8/26/2025 Initial MRP-227 augmented inspection for baffle-former bolts must be completed between 25 and 35 EFPY.

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

RFO	Cycle End Quarter/Year	Estimated EFPY	AMP-Related Scope	Inspection Method and Criteria	Comments
2R25	Spring / 2026	35.75	ASME Code Section XI ISI Initial MRP-227 augmented inspection for control rod guide tube lower flanges, upper and lower core barrel flange weld, upper and lower core barrel girth weld, and thermal shield flexures completed during or before this outage. (1) Initial MRP-227 augmented inspections for baffle-former assembly and baffle-edge bolts completed during or before this outage. Initial MRP-227 augmented inspection for internals hold down spring completed during or before this outage.	ASME Code Section XI MRP-227 inspections in accordance with MRP-228 specifications. MRP-227 inspections in accordance with MRP-228 specifications. MRP-227 inspections in accordance with MRP-228 specifications.	The initial inspection window for these components is no later than two refueling outages from the beginning of extended operation. DCPP Unit 2 has the option to perform these inspections until RO-26. The initial inspection window for the baffle-edge bolts and the baffle-former assembly is between 20 and 40 EFPY. DCPP Unit 2 has the option to perform these inspections until RO-27. The initial inspection window for the hold-down spring is within three cycles of the beginning of the license renewal period. DCPP Unit 2 has the option to perform this inspection until RO-27.
2R26	Spring / 2028	37.39			
2R27	Fall / 2029	38.94			
2R28	Spring / 2031	40.43			
2R29	Spring / 2033	42.07			
2R30	Fall / 2034	43.62	Subsequent MRP-227 augmented inspection for baffle-former bolts completed during or before this outage.(1)	MRP-227 inspections in accordance with MRP-228 specifications.	The subsequent inspection window for these components is ten years after initial inspection.

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

RFO	Cycle End Quarter/Year	Estimated EFPY	AMP-Related Scope	Inspection Method and Criteria	Comments
2R31	Spring / 2036	45.11	ASME Code Section XI ISI Subsequent MRP-227 augmented inspections for baffle-former assembly and baffle-edge bolts completed during or before this outage. (1) Subsequent MRP-227 augmented inspection for control rod guide tube lower flanges, upper and lower core barrel flange weld, upper and lower core barrel girth weld and thermal shield flexures completed during or before this outage. (1) Subsequent MRP-227 augmented inspections for baffle-former assembly and baffle-edge bolts completed during or before this outage.	MRP-227 inspections in accordance with MRP-228 specifications. MRP-227 inspections in accordance with MRP-228 specifications. MRP-227 inspections in accordance with MRP-228 specifications.	The subsequent inspection window for these components is ten years after initial inspection. The subsequent inspection window for these components is ten years after initial inspection. The subsequent inspection window for these components is ten years after initial inspection.
2R32	Spring / 2038	46.75			
2R33	Fall / 2039	48.30			
2R34	Spring / 2041	49.85			
2R35	Spring / 2043	51.49			
2R36	Fall / 2044	52.98			Renewed Operating License expires 8/26/2045.

Notes:

1. Re-examination frequency is on a 10-year basis.
2. Subsequent examination is dependent on results from the initial inspection (see Section 6 of WCAP-17451-P).

8 IMPLEMENTING DOCUMENTS

The DCPP Unit 2 AMP also references the Water Chemistry Guidelines, the ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD Program which will become part of the DCPP Unit 2 Inspection Program Plan for Reactor Vessel Internals and the Flux Thimble Tube Inspection program. MRP-227 augmented examinations (Appendix C), recommended as a result of industry programs, have been appropriately incorporated into the DCPP RVI AMP program procedure (Reference 43). The Program Plan for Inspection of Reactor Vessel Internals will coordinate with the existing DCPP Unit 2 Inservice Inspection Program and will supplement that program with the augmented examinations for managing the potential aging effects of the reactor vessel internals.

DCPP Unit 2 documents associated with the existing DCPP Unit 2 programs and considered to be implementing documents of the Reactor Vessel Internals Program are:

- Diablo Canyon Power Plant Water Chemistry Guidelines (References 15 and 16)
- Inservice Inspection Program Implementation (Reference 2)
- Flux Thimble Tube Inspection program (References 17 and 19)

The Reactor Vessel Internals Program relies on the Water Chemistry Program for maintaining high water purity to reduce susceptibility to cracking due to SCC. The Water Chemistry Plan was evaluated and found to be consistent with GALL Section XI.M2, Water Chemistry (Reference 6). Additional procedures may be updated or created as OE for augmented examinations is accumulated.

Based on this information, the updated AMP for DCPP 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. U.S Nuclear Regulatory Commission, ADAMS Accession No. ML11153A103, "Safety Evaluation Report Related to the License Renewal of Diablo Canyon Nuclear Power Plant, Units 1 and 2," Docket Nos. 50-275 and 50-323, Pacific Gas and Electric Company, June 2011.

2. Diablo Canyon Power Plant Procedure, AD5.ID2, "Inservice Inspection Program."

3. *Materials Reliability Program: Pressurized Water Reactor Internals Inspection and Evaluation Guidelines (MRP-227-A).* EPRI, Palo Alto, CA: 2011. 1022863.

4. 10 CFR Part 54, "Requirements for Renewal of Operating Licenses for Nuclear Power Plants," U.S Nuclear Regulatory Commission, July 6, 2012.

5. ASME Boiler and Pressure Vessel Code Section XI, "Rules for Inservice Inspection of Nuclear Power Plant Components" (2001 Edition through the 2003 Addenda), ASME International.

6. NUREG-1801, Rev. 2, "Generic Aging Lessons Learned (GALL) Report," U.S. Nuclear Regulatory Commission, December 2010 (updated via NRC Letter No. LR-ISG-2011-04).

7. NUREG-1800, Rev. 2, "Standard Review Plan for the Review of License Renewal Applications for Nuclear Power Plants, U.S Nuclear Regulatory Commission, December 2010.

8. Westinghouse Report WCAP-14577, Rev. 1-A, "License Renewal Evaluation: Aging Management for Reactor Internals," March 2001.

9. *Pressurized Water Reactor Primary Water Chemistry Guidelines,* Rev. 7, EPRI, Palo Alto, CA: 2014, 3002000505.

10. *Materials Reliability Program: Screening, Categorization and Ranking of PWR Internals Components for Westinghouse and Combustion Engineering PWR Design (MRP-191).* EPRI, Palo Alto, CA: 2006. 1013234.

11. *Materials Reliability Program: Inspection Standard for PWR Internals (MRP-228, Rev. 1, or current revision).* EPRI, Palo Alto, CA: 2012. 1025147.

12. Westinghouse Report, WCAP-17096-NP, Rev. 2, "Reactor Internals Acceptance Criteria Methodology and Data Requirements," December 2009.

13. "License Renewal Application: Diablo Canyon Power Plant Unit 1 and Unit 2, Facility Operating License Nos. DPR-80 and DPR-82," November 2009.

14. NEI 03-08, Rev. 2, "Guidelines for the Management of Materials Issues," Nuclear Energy Institute, Washington, DC, December 2010.

15. EPRI PWR Primary Chemistry Guidelines, Rev. 6, "Strategic Primary Water Chemistry Plan Diablo Canyon Unit 1 and Unit 2," April 2011.

16. Diablo Canyon Power Plant Program, OP F-5:I, "Chemical Control Limits and Action Guidelines for the Primary Systems."

17. Diablo Canyon Power Plant Nondestructive Examination Procedure NDE ET-9, "Eddy Current Examination of Bottom Mounted Instrumentation Flux Thimble Tubes."

18. Diablo Canyon Power Plant Surveillance Test Procedure STP R-22, "Thimble Tube inspection Program."

19. U.S. Nuclear Regulatory Commission Bulletin 88-09, "Thimble Tube Thinning in Westinghouse Reactors," July 26, 1988.

20. Diablo Canyon Action Request, A0646979, "U2: Reactor Internals Split Pin Replacement."

21. Diablo Canyon Power Plant Design Change Package, DCP N-50449, Rev. 0, "Design Change Package."

22. Diablo Canyon Power Plant UFSAR, Section 4.2.2.

23. Pacific Gas and Electric Company Program Directive OM7, "Corrective Action Program."

24. Diablo Canyon Power Plant Procedure OM7.ID1, "Problem Identification and Resolution."

25. Diablo Canyon Power Plant Procedure OM7.ID13, "Technical Evaluations."

26. Diablo Canyon Power Plant Procedure MA1.ID13, "ASME Section XI Repair/Replacement Program and Implementation."

27. Diablo Canyon Power Plant Program, OM5, "Quality Assurance Program."

28. Westinghouse Report WCAP-17451-P (Proprietary), Rev. 1, "Reactor Internals Guide Tube Wear – Westinghouse Domestic Fleet Operational Projections," October 2013.

29. U.S. Nuclear Regulatory Commission Information Notice 84-18, "Stress Corrosion Cracking in Pressurized Water Reactor Systems," March 7, 1984.

30. U.S. Nuclear Regulatory Commission Information Notice 98-11, "Cracking of Reactor Vessel Internal Baffle Former Bolts in Foreign Plants," March 25, 1998.

31. Diablo Canyon Power Plant Procedure OM4.ID3, "Operating Experience Program."

32. Diablo Canyon Power Plant Procedure XI1.DC2, "Regulatory Operating Experience (ROE)."

33. Diablo Canyon Power Plant Document, Rev. 3, "Review of Time-Limited Aging Analyses (TLAAs) for Diablo Canyon Power Plant, Units 1 and 2," February 2011.

34. PG&E Letter DCP-14-103, "10 CFR 54.21(b) Annual Update to the Diablo Canyon Power Plant License Renewal Application (LRA), Amendment 48 and LRA Appendix E, Applicant's Environment Report – Operating License Renewal Stage, Amendment 1," December 22, 2014.

35. PG&E Letter DCL-10-121, "Responses to NRC Letter dated August 25, 2010, Request for Additional Information (Set 19) for the Diablo Canyon License Renewal Application," September 22, 2010.

36. Westinghouse Letter LTR-RIAM-15-61, Rev. 0, "Summary of Diablo Canyon Units 1 and 2 Expert Elicitation Panel Meeting Minutes for Reactor Internals Components and Materials," August 25, 2015.

37. ML111990086, "NRC Regulatory Issue Summary 2011-07 License Renewal Submittal Information for Pressurized Water Reactor Internals Aging Management," U.S. Nuclear Regulatory Commission, July 2011.

38. License Renewal Issue No. 98-0030, "Thermal Aging Embrittlement of Cast Austenitic Stainless Steel Components," U.S. Nuclear Regulatory Commission, May 19, 2000 (NRC ADAMS Accession No. ML003717179).

39. NUREG/CR-4513, ANL-93/22, Rev. 1, "Estimation of Fracture Toughness of Cast Stainless Steels During Thermal Aging in LWR Systems," U.S. Nuclear Regulatory Commission, May 1994 (NRC ADAMS Accession No. ML052360554).

40. ASME Boiler and Pressure Vessel Code, Section II, 1968 Edition.

41. ASME Boiler and Pressure Vessel Code, Section II, 2013 Edition.

42. EPRI Letter, MRP 2014-006, *Materials Reliability Program:* Pressurized Water Reactor Internals Inspection and Evaluation Guidelines (MRP-227-A), EPRI, Palo Alto, CA: 2011. 10122863, Transmittal of Interim Guidance, February 18, 2014.

43. Diablo Canyon Power Plant Procedure TS1.ID11, Reactor Internals Aging Management Program.

APPENDIX A
ILLUSTRATIONS

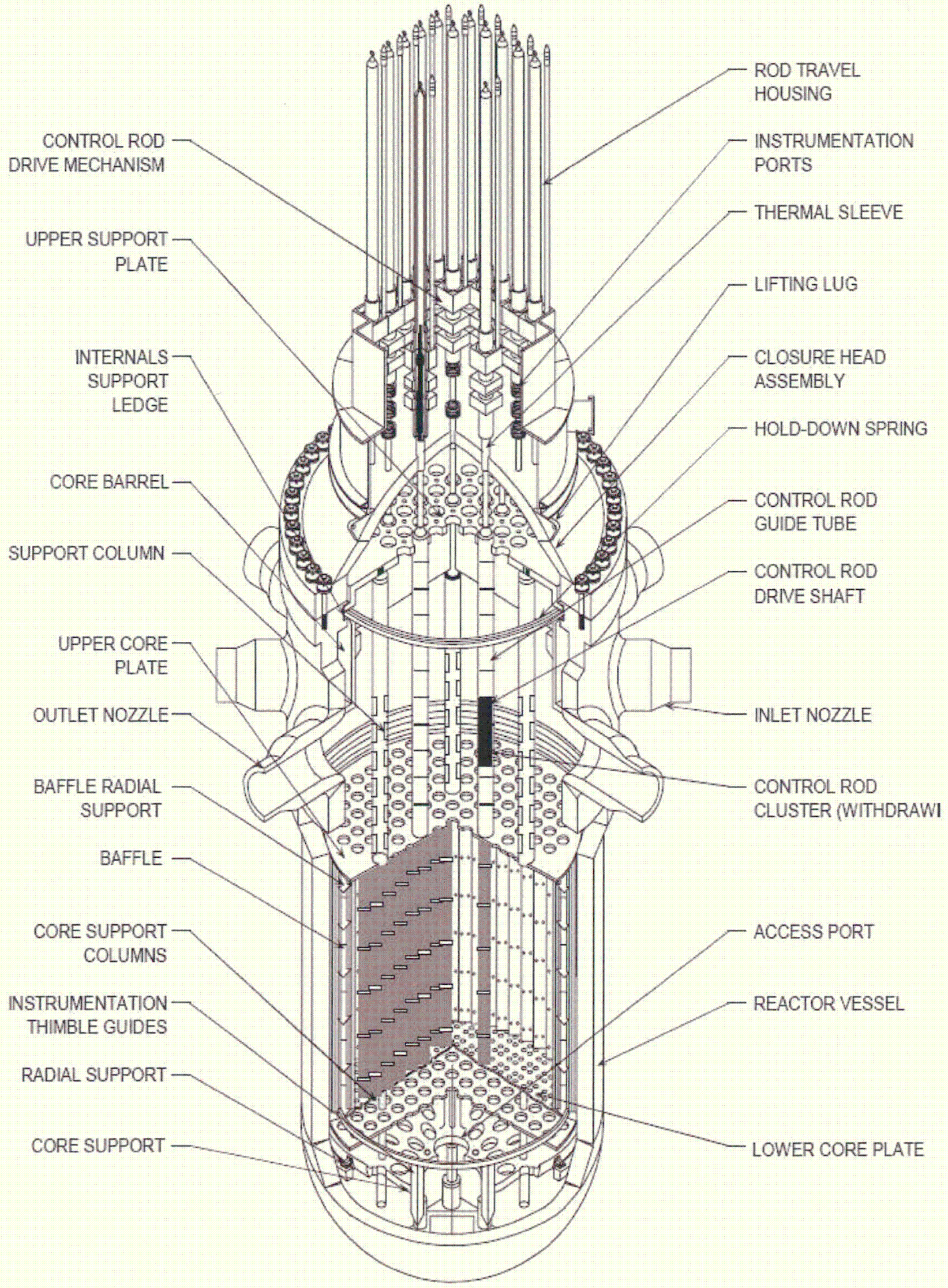

Figure A-1 Illustration of Typical Westinghouse Internals

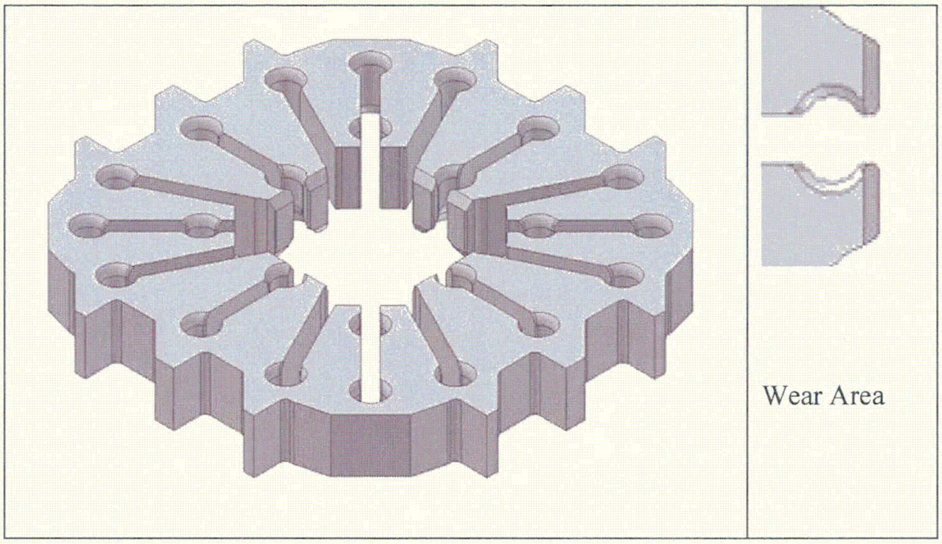

Figure A-2 Typical Westinghouse Control Rod Guide Card

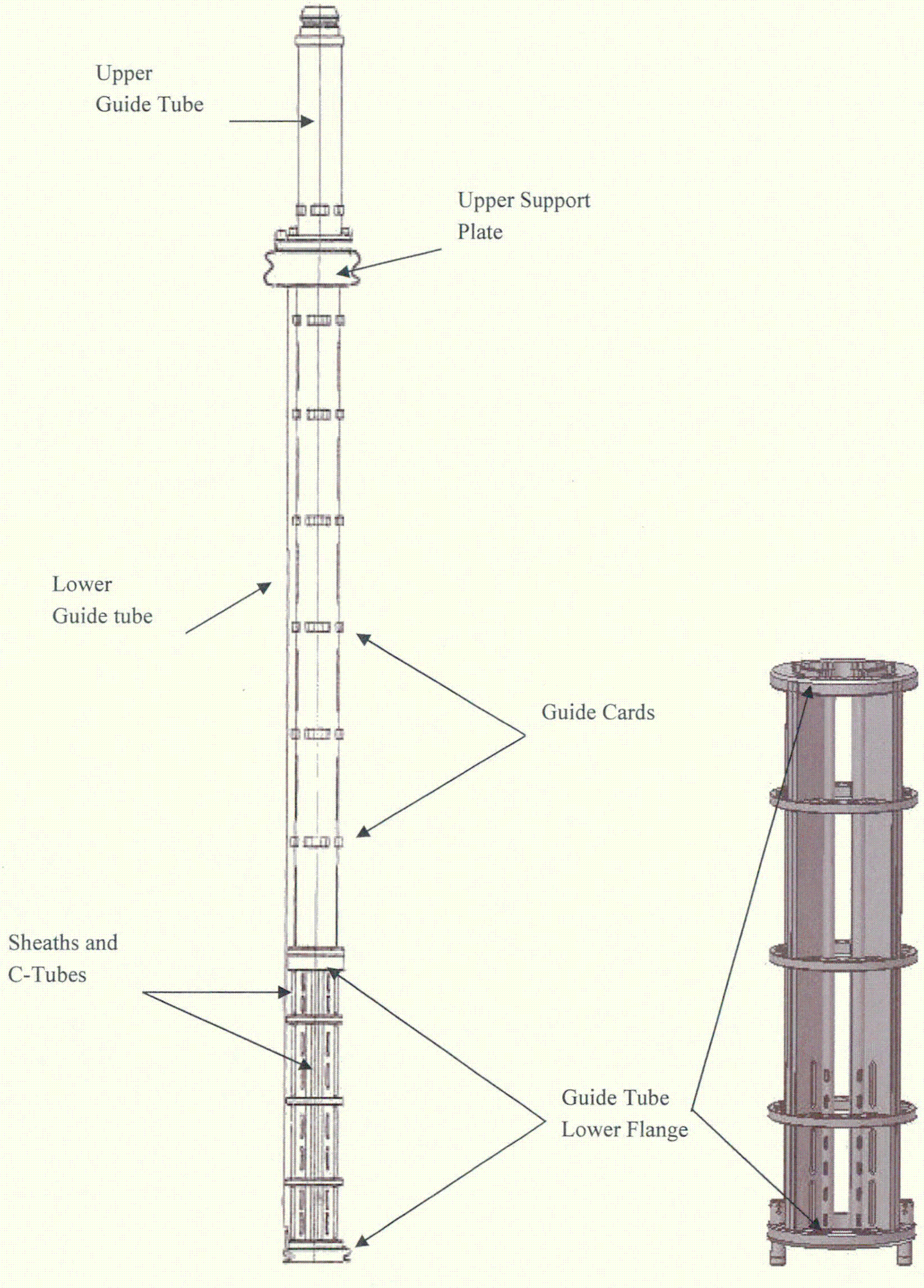

Figure A-3 Typical Lower Section of Control Rod Guide Tube Assembly

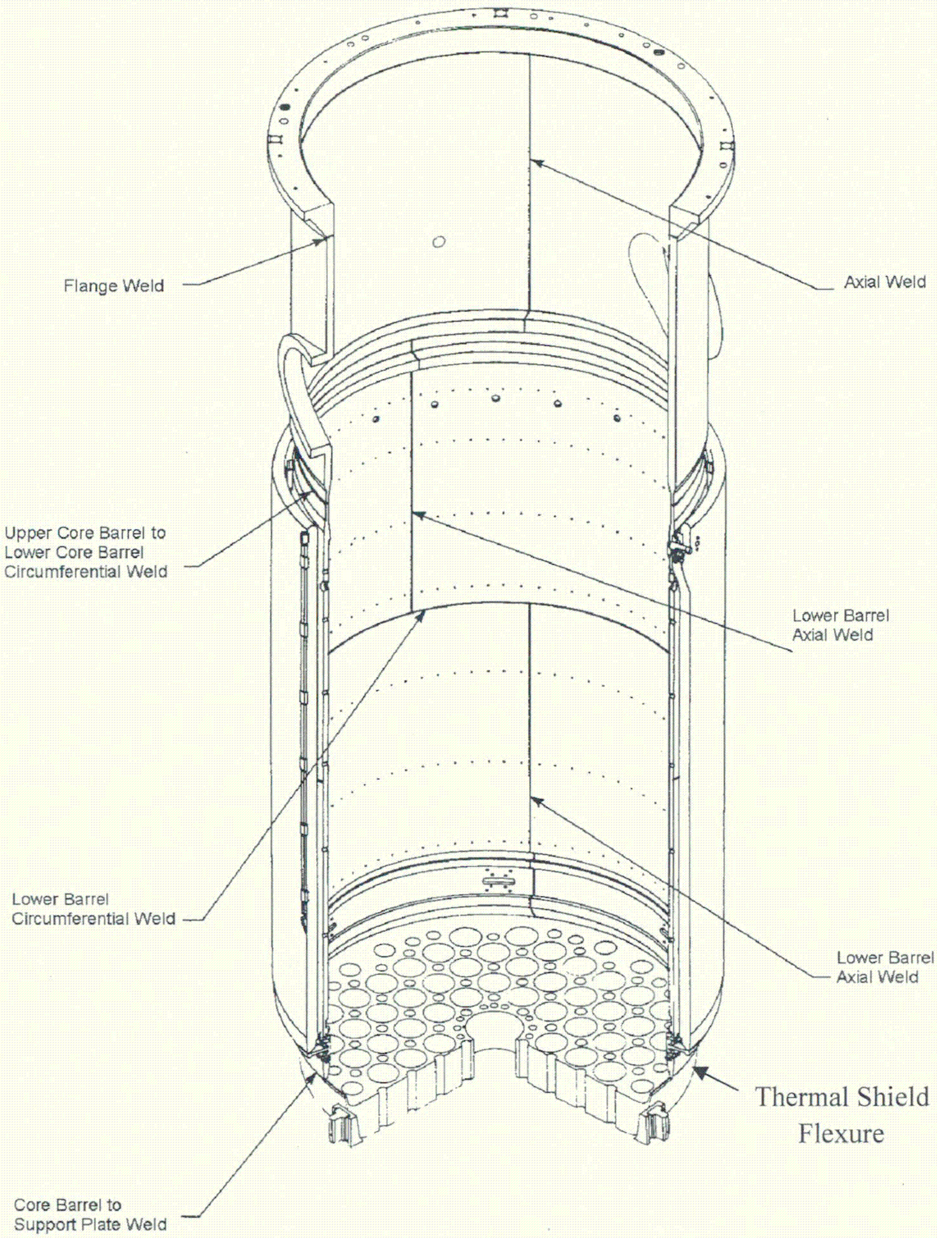

Figure A-4 Major Core Barrel Welds
(Note: Thermal shield flexure does not exist at DCPP Unit 2.)

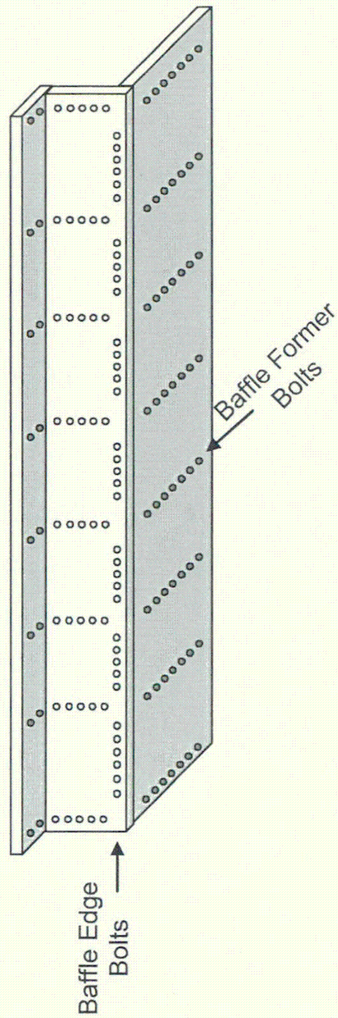

Figure A-5 Bolting Systems Used in Westinghouse Core Baffles

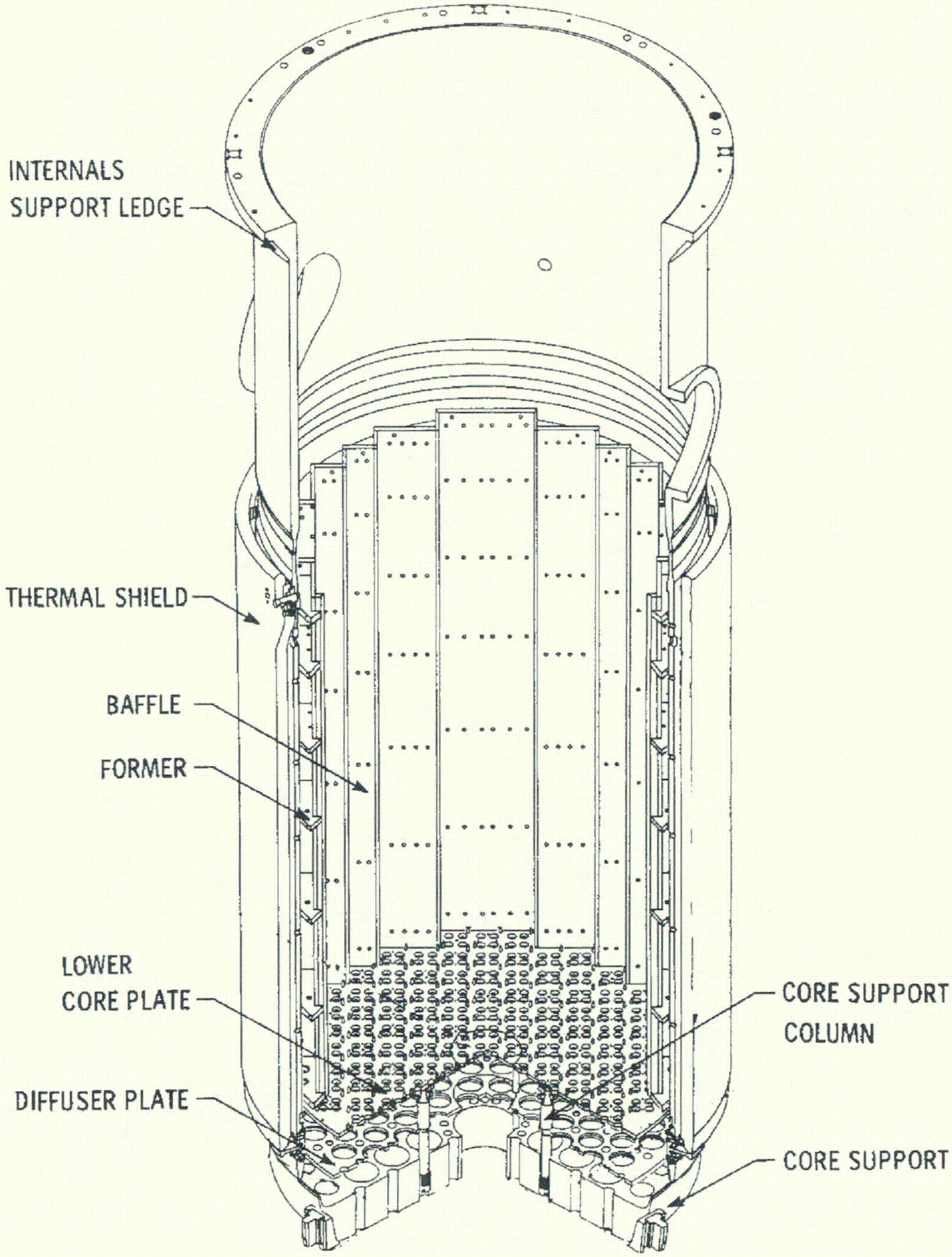

Figure A-6 Core Baffle/Barrel Structure
(Note: Thermal shield flexure does not exist at DCPP Unit 2.)

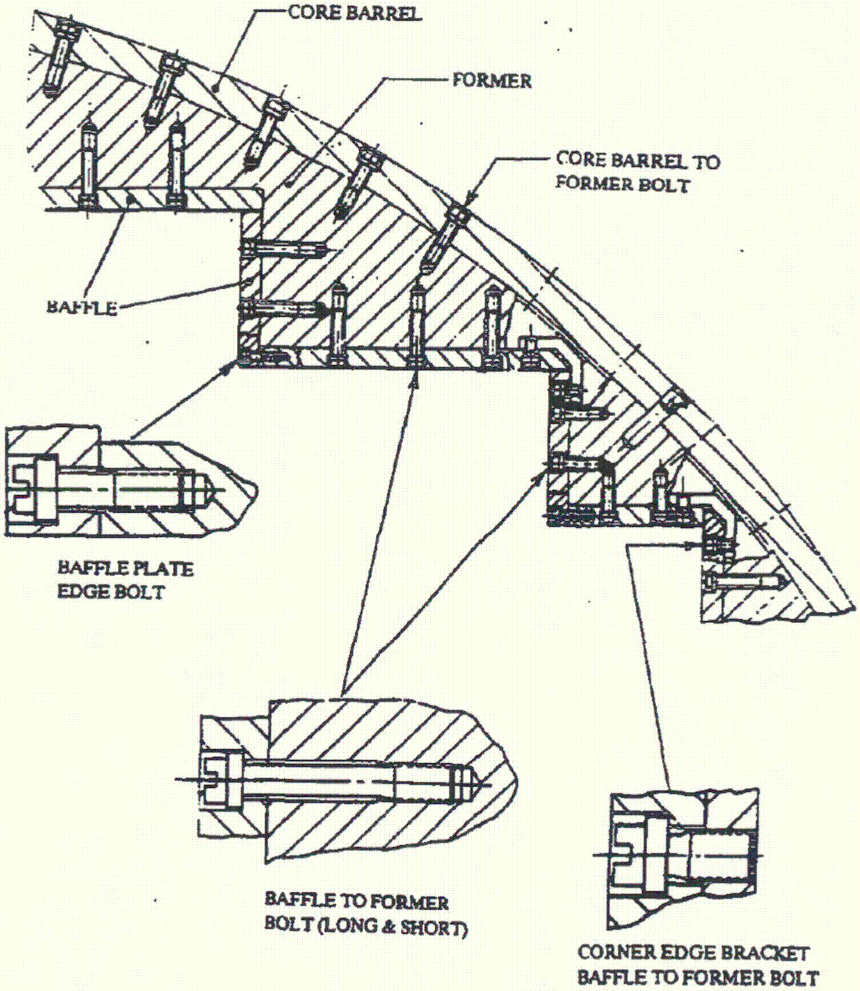

Figure A-7 Bolting in a Typical Westinghouse Baffle-Former Structure

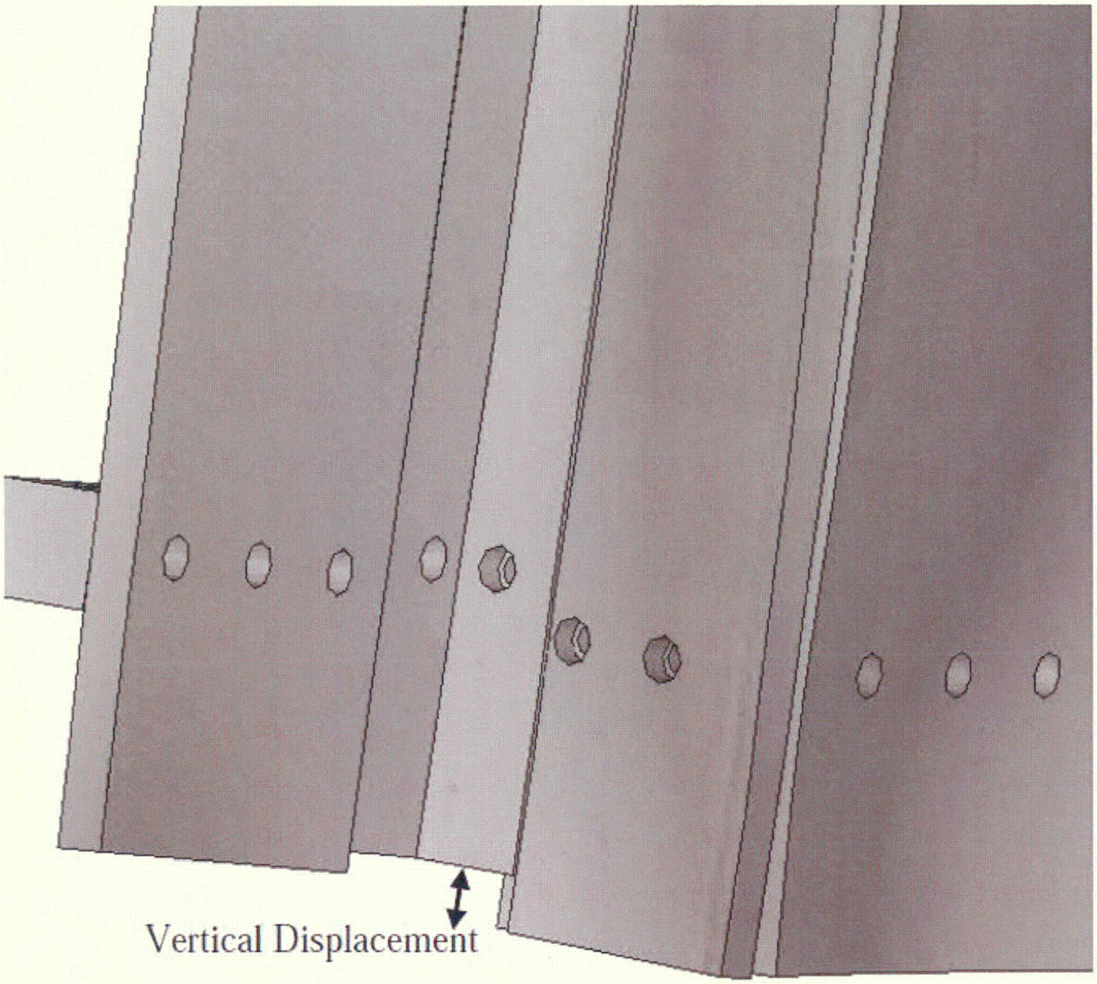

Figure A-8 Vertical Displacement between the Baffle Plates and Bracket at the Bottom of the Baffle-Former-Barrel Assembly (exaggerated)

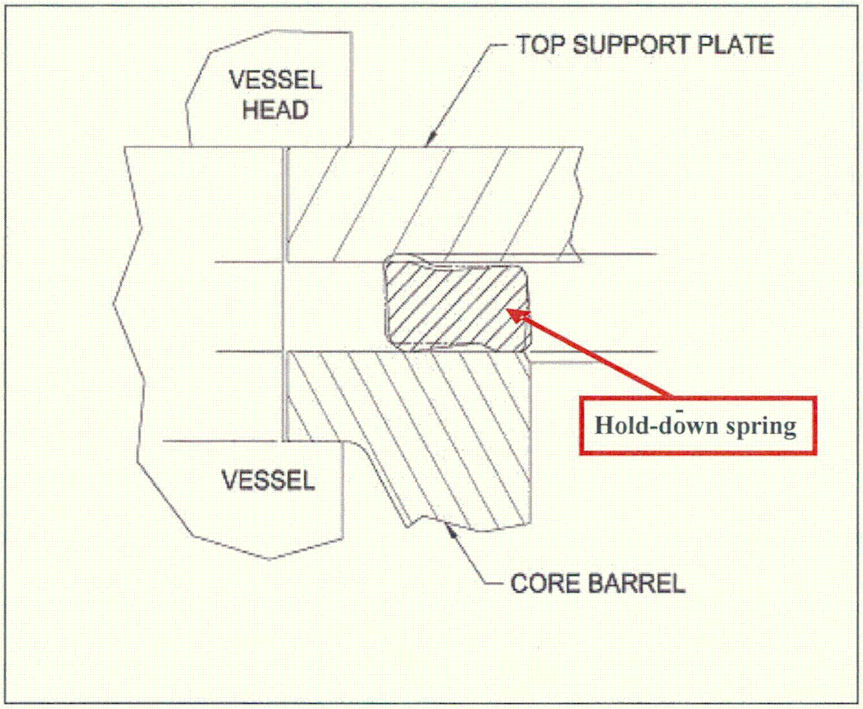

Figure A-9 Schematic Cross-Sections of the Westinghouse Hold-Down Springs

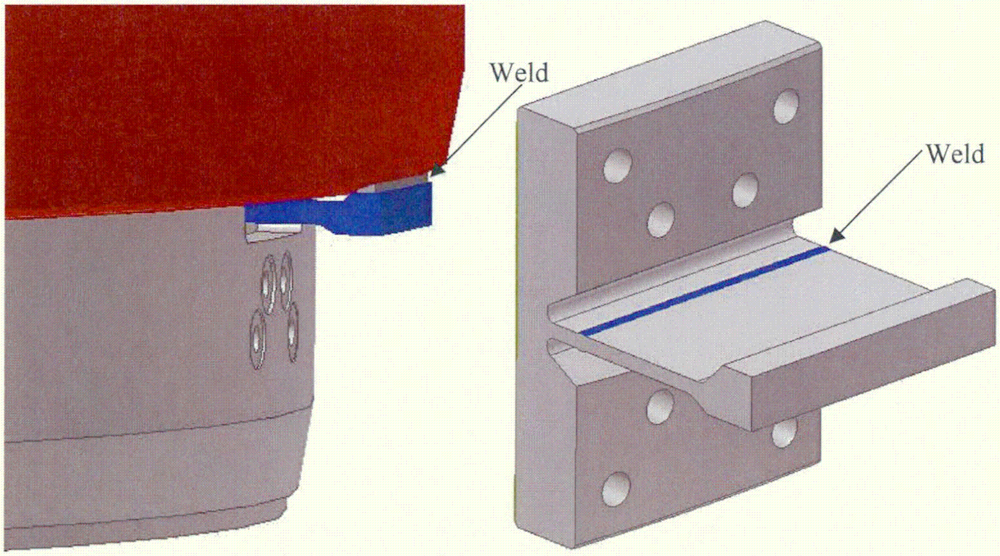

Figure A-10 Typical Thermal Shield Flexure
(Note: Not applicable to DCPP Unit 2)

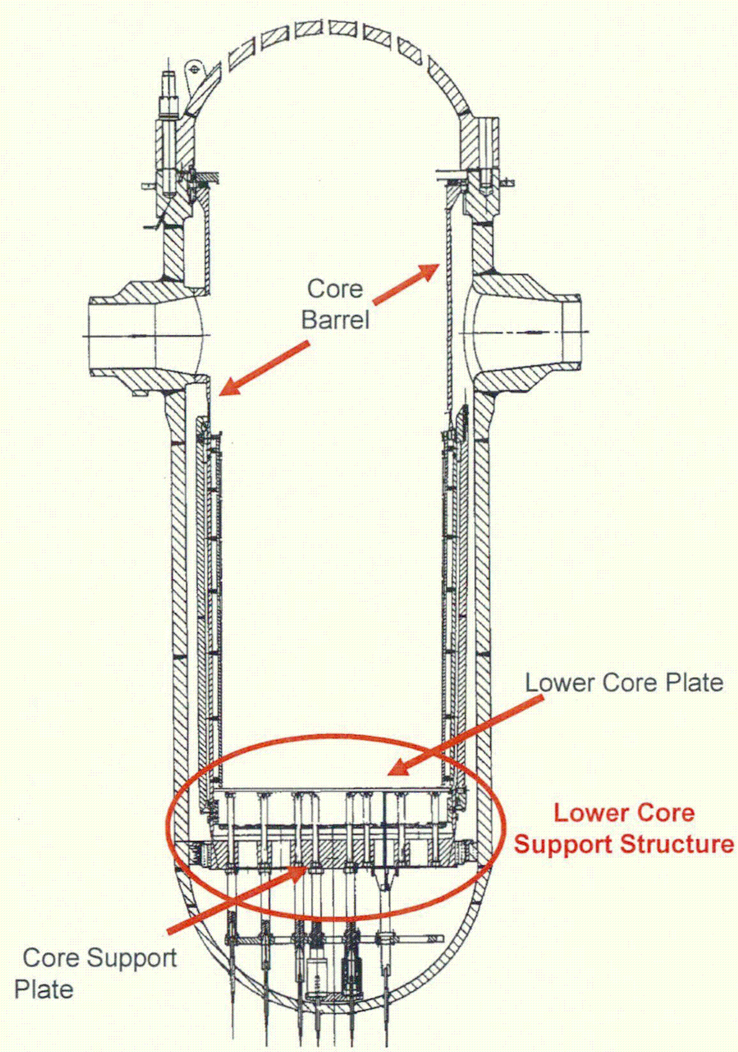

Figure A-11 Lower Core Support Structure

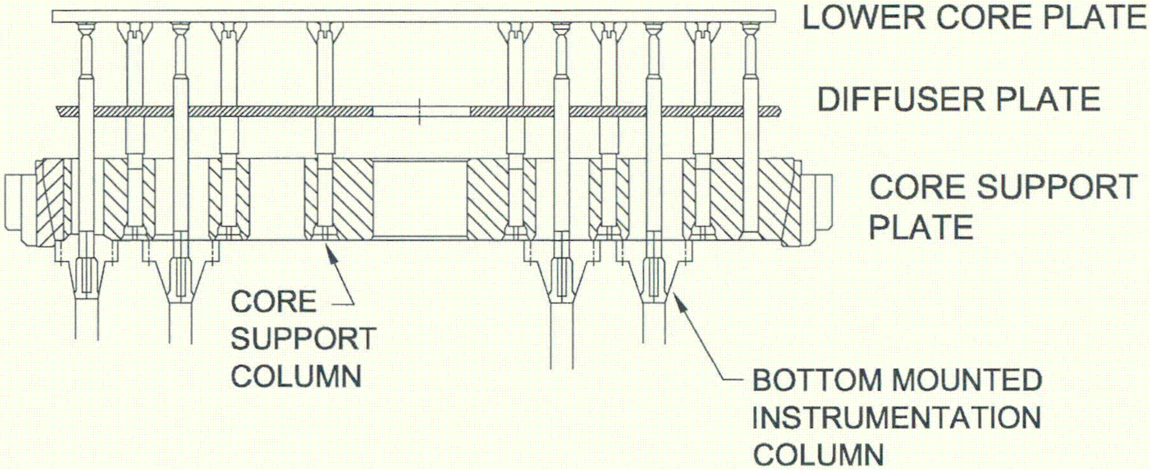

Figure A-12 Lower Core Support Structure – Core Support Plate Cross-Section

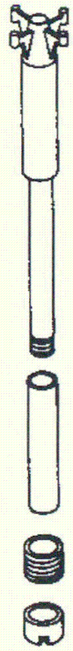

Figure A-13 Typical Core Support Column

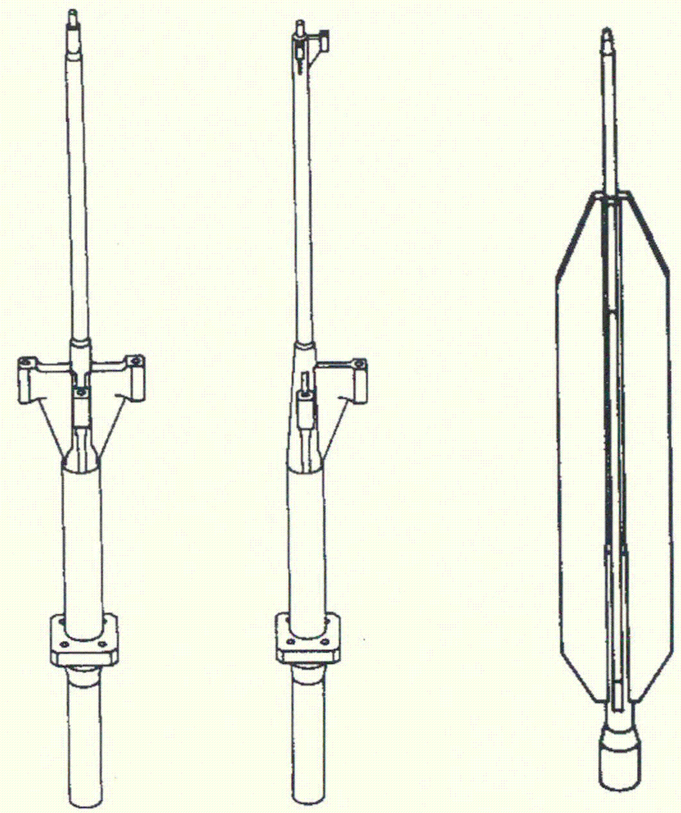

Figure A-14 Examples of Bottom-Mounted Instrumentation (BMI) Column Designs

APPENDIX B
DIABLO CANYON POWER PLANT LICENSE RENEWAL AGING MANAGEMENT REVIEW SUMMARY TABLES

The content and numerical identifiers in Table B-1 are extracted from Table 3.1.2-1 "Reactor Vessel Internals (Westinghouse) Summary of Aging Management Evaluation" of the Diablo Canyon Power Plant LRA.

Table B-1 LRA Aging Management Evaluation Summary—Table 3.1.2-1 of the DCPP LRA

Component Type	Aging Effect Requiring Management	Aging Management Program	NUREG-1801, Vol. 2 Item	LRA Table 3.1.1 Item #	Notes
RVI Control Rod Guide Tube Assembly (Guide Tube Bolts)	Loss of Material	ASME Section XI Inservice Inspection, Subsections IWB, IWC and IWD (B2.1.1)	IV.B2.BP-382	3.1.1.63	A, 3
	Cracking	ASME Section XI Inservice Inspection, Subsections IWB, IWC and IWD (B2.1.1)	IV.B2.BP-382	3.1.1.63	A, 3
RVI Core Barrel Assembly (Core Barrel Flange)	Loss of Material	ASME Section XI Inservice Inspection, Subsections IWB, IWC and IWD (B2.1.1)	IV.B2.RP-382	3.1.1.63	A, 3
RVI Core Barrel Assembly (Core Barrel Outlet Nozzles)	Cracking	ASME Section XI Inservice Inspection, Subsections IWB, IWC and IWD (B2.1.1)	IV.B2.RP-382	3.1.1.63	A, 3
	Loss of Material	ASME Section XI Inservice Inspection, Subsections IWB, IWC and IWD (B2.1.1)	IV.B2.RP-382	3.1.1.63	A, 3
RVI Core Barrel Assembly (Core Barrel Outlet Nozzle Welds)	Cracking	Water Chemistry (B2.1.2) and PWR Vessel Internals (B2.1.41)	IV.B2.RP-278	3.1.1.80	A, 3

Table B-1 LRA Aging Management Evaluation Summary—Table 3.1.2-1 of the DCPP LRA

Component Type	Aging Effect Requiring Management	Aging Management Program	NUREG-1801, Vol. 2 Item	LRA Table 3.1.1 Item #	Notes
	Loss of Fracture Toughness	PWR Vessel Internals (B2.1.41)	IV.B2.RP-278a	3.1.1.27	A, 3
RVI Core Barrel Assembly (Upper and Lower Core Barrel Girth Welds)	Cracking	Water Chemistry (B2.1.2) and PWR Vessel Internals (B2.1.41)	IV.B2.RP-387	3.1.1.30	A, 3
RVI Core Barrel Assembly (Upper and Lower Core Barrel Axial Welds)	Cracking	Water Chemistry (B2.1.2) and PWR Vessel Internals (B2.1.41)	IV.B2.RP-387a	3.1.1.80	A, 3
	Loss of Fracture Toughness	PWR Vessel Internals (B2.1.41)	IV.B2.RP-388a	3.1.1.27	A, 3
RVI Core Barrel Assembly (Lower Core Barrel Flange Weld)	Cracking	Water Chemistry (B2.1.2) and PWR Vessel Internals (B2.1.41)	IV.B2.RP-280	3.1.1.30	A, 3
RVI Core Barrel Assembly (Upper Core Barrel Flange Weld)	Cracking	Water Chemistry (B2.1.2) and PWR Vessel Internals (B2.1.41)	IV.B2.RP-276	3.1.1.30	A, 3
RVI Baffle-to-Former Assembly (Baffle Plates, Former Plates)	Changes in Dimensions	PWR Vessel Internals (B2.1.41)	IV.B2.RP-270	3.1.1.33	A, 3
	Cracking	Water Chemistry (B2.1.2) and PWR Vessel Internals (B2.1.41)	IV.B2.RP-270a	3.1.1.30	A, 3
	Loss of Material	Water Chemistry (B2.1.2)	IV.B2-32	3.1.1.83	A
RVI Baffle-to-Former Assembly (Baffle/ Former Bolts)	Changes in Dimensions	PWR Vessel Internals (B2.1.41)	IV.B2.RP-272	3.1.1.33	A, 3
	Loss of Preload	PWR Vessel Internals (B2.1.41)	IV.B2.RP-272	3.1.1.33	A, 3
	Loss of Fracture Toughness	PWR Vessel Internals (B2.1.41)	IV.B2.RP-272	3.1.1.33	A, 3

Table B-1 LRA Aging Management Evaluation Summary—Table 3.1.2-1 of the DCPP LRA

Component Type	Aging Effect Requiring Management	Aging Management Program	NUREG-1801, Vol. 2 Item	LRA Table 3.1.1 Item #	Notes
RVI Baffle-to-Former Assembly (Baffle/ Former Bolts)	Cracking	Water Chemistry (B2.1.2) and PWR Vessel Internals (B2.1.41)	IV.B2.RP-271	3.1.1.30	A, 3
	Loss of Material	Water Chemistry (B2.1.2)	IV.B2-32	3.1.1.83	A
RVI Baffle-to-Former Assembly (Baffle-Edge bolts)	Cracking	Water Chemistry (B2.1.2) and PWR Vessel Internals (B2.1.41)	IV.B2.RP-275	3.1.1.30	A, 3
	Loss of Fracture Toughness	PWR Vessel Internals (B2.1.41)	IV.B2.RP-354	3.1.1.33	A, 3
	Loss of Preload	PWR Vessel Internals (B2.1.41)	IV.B2.RP-354	3.1.1.33	A, 3
	Changes in Dimension	PWR Vessel Internals (B2.1.41)	IV.B2.RP-354	3.1.1.33	A, 3
RVI Lower Core Support Structure (Lower Core Plate)	Loss of Fracture Toughness	PWR Vessel Internals (B2.1.41)	IV.B2.RP-288	3.1.1.22	A, 3
	Loss of Material	PWR Vessel Internals (B2.1.41)	IV.B2.RP-288	3.1.1.22	A, 3
	Cracking	Water Chemistry (B2.1.2) and PWR Vessel Internals (B2.1.41)	IV.B2.RP-289	3.1.1.37	A, 3
RVI Lower Core Support Structure (Lower Support Forging of Casting)	Loss of fracture toughness	PWR Vessel Internals (B2.1.41)	IV.B2.RP-290a	3.1.1.27	A, 3
	Cracking	Water Chemistry (B2.1.2) and PWR Vessel Internals (B2.1.41)	IV.B2.RP-291a	3.1.1.80	A, 3
RVI Lower Core Support Structure	Loss of preload	PWR Vessel Internals (B2.1.41)	IV.B2.RP-285	3.1.1.22	A, 3

Table B-1 LRA Aging Management Evaluation Summary—Table 3.1.2-1 of the DCPP LRA

Component Type	Aging Effect Requiring Management	Aging Management Program	NUREG-1801, Vol. 2 Item	LRA Table 3.1.1 Item #	Notes
(Clevis Insert Bolts)	Loss of Material	PWR Vessel Internals (B2.1.41)	IV.B2.RP-285	3.1.1.22	A, 3
	Cracking	Water Chemistry (B2.1.2) and PWR Vessel Internals (B2.1.41)	IV.B2.RP-399	3.1.1.37	A, 3
RVI Lower Core Support Structure (Radial Keys, Clevis Insert Keyways)	Cumulative Fatigue Damage	Time-Limited Aging Analysis evaluated for the period of extended operation	IV.B2.RP-382	3.1.1.63	C, 3
	Cracking	ASME Section XI Inservice Inspection, Subsections IWB, IWC and IWD (B2.1.1)	IV.B2.RP-382	3.1.1.63	C, 3
	Loss of Material	ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD (B2.1.1)	IV.B2-RP-382	3.1.1.63	C, 3
RVI Lower Core Support Structure (Core Support Column Bolts)	Cracking	Water Chemistry (B2.1.2) and PWR Vessel Internals (B2.1.41)	IV.B2.RP-286	3.1.1.80	A, 3
	Loss of Fracture Toughness	PWR Vessel Internals (B2.1.41)	IV.B2.RP-287	3.1.1.27	A, 3
	Loss of Preload	PWR Vessel Internals (B2.1.41)	IV.B2.RP-287	3.1.1.27	A, 3
RVI Lower Core Support Structure (Lower Support Column Bodies (non-cast))	Cracking	Water Chemistry (B2.1.2) and PWR Vessel Internals (B2.1.41)	IV.B2.RP-294	3.1.1.80	A, 3
	Loss of Fracture Toughness	PWR Vessel Internals (B2.1.41)	IV.B2.RP-295	3.1.1.27	A, 3

Table B-1 LRA Aging Management Evaluation Summary—Table 3.1.2-1 of the DCPP LRA

Component Type	Aging Effect Requiring Management	Aging Management Program	NUREG-1801, Vol. 2 Item	LRA Table 3.1.1 Item #	Notes
RVI Lower Core Support Structure (Lower Support Column Bodies [cast])	Cracking	Water Chemistry (B2.1.2) and PWR Vessel Internals (B2.1.41)	IV.B2.RP-291	3.1.1.80	A, 3
	Loss of Facture toughness	PWR Vessel Internals (B2.1.41)	IV.B2.RP-290	3.1.1.27	A, 3
RVI Lower Core Support Structure (All RVI Stainless Steel Components)	Cumulative Fatigue Damage	Time-Limited Aging Analysis evaluated for the period of extended operation	IV.B2-31	3.1.1.05	A
	Loss of Material	Water Chemistry (B2.1.2)	IV.B2-32	3.1.1.83	A
RVI Lower Core Support Structure (Core Support Casting [U1])	Cumulative Fatigue Damage	Time-Limited Aging Analysis evaluated for the period of extended operation	IV.B2-31	3.1.1.05	A
	Loss of Material	Water Chemistry (B2.1.2)	IV.B2-32	3.1.1.83	A
RVI Thermal Shield Assembly (Thermal Shield Flexures)	Loss of material	PWR Vessel Internals (B2.1.41)	IV.B2.RP-302a	3.1.1.33	A, 3
	Cracking	Water Chemistry (B2.1.2) and PWR Vessel Internals (B2.1.41)	IV.B2.RP-302	3.1.1.30	A, 3
RVI Upper Core Support Structure (Upper Core Plate)	Cumulative Fatigue Damage	Time-Limited Aging Analysis evaluated for the period of extended operation	IV.B2-31	3.1.1.05	A
	Loss of Material	PWR Vessel Internals (B2.1.41)	IV.B2.RP-290b	3.1.1.27	A, 3
	Cracking	PWR Vessel Internals (B2.1.41)	IV.B2.RP-291b	3.1.1.80	A, 3

Table B-1 LRA Aging Management Evaluation Summary—Table 3.1.2-1 of the DCPP LRA

Component Type	Aging Effect Requiring Management	Aging Management Program	NUREG-1801, Vol. 2 Item	LRA Table 3.1.1 Item #	Notes
RVI Upper Core Support Structure (Upper Core Plate Alignment Pins)	Loss of Material	PWR Vessel Internals (B2.1.41)	IV.B2.RP-299	3.1.1.22	A, 3
	Cracking	Water Chemistry (B2.1.2) and PWR Vessel Internals (B2.1.41)	IV.B2.RP-301	3.1.1.37	A, 3
RVI Upper Core Support Structure (Upper Support Columns)	Cumulative Fatigue Damage	Time-Limited Aging Analysis evaluated for the period of extended operation	IV.B2-31	3.1.1.05	A
	Loss of Material	Water Chemistry (B2.1.2)	IV.B2-32	3.1.1.83	A
RVI Upper Core Support Structure (Upper Support Ring or Skirt)	Cracking	Water Chemistry (B2.1.2) and PWR Vessel Internals (B2.1.41)	IV.B2.RP-346	3.1.1.37	A, 3
RVI Hold-Down Spring (RVI Hold-Down Spring)	Loss of Material	Water Chemistry (B2.1.2)	IV.B2.RP-300	3.1.1.33	A, 3
	Loss of Preload	PWR Vessel Internals (B2.1.41)	IV.B2.RP-300	3.1.1.33	A, 3
	Changes in Dimensions	PWR Vessel Internals (B2.1.41)	IV.B2.RP-300	3.1.1.33	A, 3
RVI Control Rod Guide Tube Assembly (Control Rod Guide Tubes/Tube Support Pins/Guide Plates [Cards])	Loss of Material	PWR Vessel Internals (B2.1.41)	IV.B2.RP-296	3.1.1.33	A, 3
RVI Control Rod Guide Tube Assembly (Lower Flange Welds)	Loss of Fracture Toughness	PWR Vessel Internals (B2.1.41)	IV.B2.RP-297	3.1.1.33	A, 3

Table B-1 LRA Aging Management Evaluation Summary—Table 3.1.2-1 of the DCPP LRA

Component Type	Aging Effect Requiring Management	Aging Management Program	NUREG-1801, Vol. 2 Item	LRA Table 3.1.1 Item #	Notes
RVI Components	Loss of Material	Water Chemistry (B2.1.2)	IV.B2-32	3.1.1.83	A
RVI Control Rod Guide Tube Assembly (Guide Tube Bolts)	Cracking	ASME Section XI Inservice Inspection, Subsections IWB, IWC and IWD (B2.1.1)	IV.B2.BP-382	3.1.1.63	A, 3
	Loss of Material	ASME Section XI Inservice Inspection, Subsections IWB, IWC and IWD (B2.1.1)	IV.B2.BP-382	3.1.1.63	A, 3
RVI Baffle-to-Former Assembly (Barrel-to-Former Bolts)	Changes in Dimensions	PWR Vessel Internals (B2.1.41)	IV.B2.RP-274	3.1.1.27	A, 3
	Loss of Preload	PWR Vessel Internals (B2.1.41)	IV.B2.RP-274	3.1.1.27	A, 3
	Cracking	Water Chemistry (B2.1.2) and PWR Vessel Internals (B2.1.41)	IV.B2.RP-273	3.1.1.80	A, 3
	Loss of Fracture Toughness	PWR Vessel Internals (B2.1.41)	IV.B2.RP-274	3.1.1.27	A, 3
BMI System (BMI Column Bodies)	Loss of Fracture Toughness	PWR Vessel Internals (B2.1.41)	IV.B2.RP-292	3.1.1.27	A, 3
	Cracking	Water Chemistry (B2.1.2) and PWR Vessel Internals	IV.B2.RP-293	3.1.1.80	A, 3

Standard Notes:

A Consistent with NUREG-1801 item for component, material, environment, and aging effects. AMP is consistent with NUREG-1801 AMP.

C Component is different, but consistent with NUREG-1801 item for material, environment, and aging effect. AMP is consistent with NUREG-1801 AMP.

E Consistent with NUREG-1801 for material, environment, and aging effect, but a different aging management program is credited or NUREG-1801 identifies a plant-specific aging management program.

Plant-Specific Note:

3 Line item was revised to align with NUREG-1801, Revision 2 (Reference 6) and LR-ISG-2011-04. Reference PG&E Letter DCL-14-103 (Reference 34)

APPENDIX C
MRP-227 AUGMENTED INSPECTIONS

Table C-1 MRP-227 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(7). (Reference 42)	Minimum examination of 20% of the number of CRGT assemblies, and as per the requirements of WCAP-17451-P Revision 1 Section 5(7).See Figure A-2. (Reference 42)
Control Rod Guide Tube Assembly Lower flange welds	All plants	Cracking (SCC, Fatigue) Aging Management (IE and TE)	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 periphery CRGT assemblies(2). See Figure A-3.
Core Barrel Assembly Upper core barrel flange weld	All plants	Cracking (SCC)	Remaining core barrel welds, Lower support column bodies (non-cast)	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(4). See Figure A-4.

Table C-1 MRP-227 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
Core Barrel Assembly Upper and lower core barrel cylinder girth welds	All plants	Cracking (SCC, IASCC, Fatigue)	None	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(4).
Core Barrel Assembly Lower core barrel flange weld(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 examination on a ten-year interval.	100% of one side of the accessible surfaces of the selected weld and adjacent base metal(4).

Table C-1 MRP-227 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-edge bolts	All plants with baffle-edge bolts (Applicable to DCPP 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)(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(3). See Figures A-5, A-6, and A-7.
Baffle-Former Assembly Baffle-former bolts	All plants	Cracking (IASCC, Fatigue)	Lower support column bolts, barrel-former bolts	Baseline volumetric (UT) examination between 25 and 35 EFPY, with subsequent examination after 10 to 15 additional EFPY to confirm stability of bolting pattern. Re-examination for high-leakage core designs requires continuing examinations on a ten-year interval.	100% of accessible bolts(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-6.

Table C-1 MRP-227 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 Assembly	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 joints	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 Figure A-5 and A-8.

Table C-1 MRP-227 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
Alignment and Interfacing Components Internals hold-down spring	All plants with 304 stainless steel hold-down springs Note: DCPP Unit 2 hold-down spring is Type 403 SS. (Not applicable to DCPP Unit 2)	Distortion (Loss of Load) Note: This mechanism was not strictly identified in the original list of age-related degradation mechanisms (ARDM).	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. Replacement of 304 springs by 403 springs is required when the spring stiffness is determined to relax beyond design tolerance. See Figure A-9.
Thermal Shield Assembly Thermal shield flexures	All plants with thermal shields (Not applicable to DCPP 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 MRP-227 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

Note:
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 Table 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 Table 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 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.

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

Item	Applicability	Effect (Mechanism)	Primary Link (Note 1)	Examination Method (Note 1)	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(2).
Lower Internals Assembly Lower Support Forging or Castings	All plants	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(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(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(2). See Figures A-11, A-12 and A-13.
Core Barrel Assembly Core barrel outlet nozzles	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(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(2). See Figure A-4.

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

Item	Applicability	Effect (Mechanism)	Primary Link (Note 1)	Examination Method (Note 1)	Examination Coverage
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(2). See Figures A-11, 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(2). See Figures A-11, A-12, and A-13.
BMI System 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. Flux thimble insertion/withdrawal to be monitored at each inspection interval. Re-inspection every 10 years following initial inspection.	100% of BMI column bodies for which difficulty is detected during flux thimble insertion/withdrawal. See Figures A-12 and A-14.

Note:

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, 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).

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

Item	Applicability	Effect (Mechanism)	Primary Link	Examination Method/Frequency	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(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(1)	All plants	Loss of material (Wear)	ASME Code Section XI	Visual (VT-3) examination.	All accessible surfaces at specified frequency.
BMI 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) (2)	ASME Code Section XI	Visual (VT-3) examination.	All accessible surfaces at specified frequency.
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, which is not applicable at DCPP Unit 2.
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 Acceptance Criteria and Expansion Criteria Recommendations for Westinghouse-Designed Internals

Item	Applicability	Examination Acceptance Criteria(1)	Expansion Link(s)	Expansion Criteria	Additional Examination Acceptance Criteria
Control Rod Guide Tube Assembly Guide plates (cards)	All plants	Visual (VT-3) Examination(2) 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
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. 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 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 MRP-227 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
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, and any supplementary UT examination, be expanded to include the core outlet nozzle welds by the completion of the next refueling outage. b. If extensive cracking in the remaining core barrel 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 is a detectable crack-like surface indication.
Core Barrel Assembly Lower core barrel flange weld(2)	All plants	Periodic enhanced visual (EVT-1) examination. The specific relevant condition is a detectable crack-like surface indication.	None	None	None

Table C-4 MRP-227 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
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.
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 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.
Baffle-Former Assembly Baffle-edge bolt	All plants with baffle-edge bolts (Applicable to DCPP 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 MRP-227 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
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 MRP-227 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
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 MRP-227 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
Alignment and Interfacing Components Internals hold down spring	All plants with 304 stainless steel hold down springs Note: DCPP Unit 2 hold down spring is Type 403 SS. (Not applicable to DCPP Unit 2)	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
Thermal Shield Assembly Thermal shield flexures	All plants with thermal shields (Not applicable for DCPP 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

Note:
1. The examination acceptance criterion for visual examination is the absence of the specified relevant condition(s).
2. WCAP-17451-P Revision 1 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 level.