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Materials Reliability Program: Failure Modes, Effects, and Criticality Analysis of B&W-Designed PWR Internals (MRP-190)

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Technical Report

Materials Reliability Program: Failure Modes, Effects, and Criticality Analysis of B&W-Designed PWR Internals (MRP-190)

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Technical Report, November 2006

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REPORT SUMMARY

Management of aging effects—such as loss of material, reduction in fracture toughness, or cracking—depends on the demonstrated capability to detect, evaluate, and potentially correct conditions that could affect system, structure, or component function. This report presents results of the failure modes, effects, and criticality analysis (FMECA) of Babcock & Wilcox (B&W)-designed pressurized water reactor (PWR) internals. Results from the FMECA help provide the technical bases for screening, ranking, and categorization for age-related degradation mechanisms of PWR internals component items.

Background

The framework for implementation of an aging management program for PWR internals component items, using inspections and flaw tolerance evaluations to manage age-related degradation issues, has been developed and is documented in EPRI Materials Reliability Program (MRP) reports MRP-134 and MRP-153, cited in the EPRI Perspective. Important elements of this framework include

- Screening, categorizing, and ranking of PWR internals component items for susceptibility and significance to age-related degradation mechanisms
- Functionality analyses and safety assessment of PWR internals component items to define a safe and cost-effective aging management in-service inspection and evaluation method and strategy

This report documents the development and evaluation results of an FMECA, performed by AREVA NP, to help provide a technical basis for screening, ranking, and categorizing age-related degradation mechanisms of B&W-designed PWR internals component items.

Objectives

To provide a systematic, semi-quantitative analysis of the B&W-designed PWR internals component items in order to identify combinations of internals component items and age-related degradation mechanisms that potentially result in degradation leading to significant safety or economic risk.

Approach

The research team first identified all B&W-designed PWR internals component items and developed FMECA tables for each one, considering degradation mechanism, failure mode, local failure effects, global failure effects, criticality metrics (susceptibility and severity of consequence), and failure mode detectibility. Subsequently, the FMECA tables were populated through an expert panel elicitation process. Finally, the team developed a risk matrix to correlate the consequence severity of a particular failure mode with the susceptibility of a particular age-related degradation mechanism occurring. Different risk bands were used within the matrix to

categorize the level of safety or economic risk of a particular component item/degradation mechanism pair, where risk is defined as the likelihood (susceptibility of an event) times the consequence (severity of the event).

Results

FMECA tables were developed for the following four B&W-designed PWR internals assemblies, including the

- Plenum assembly
- Core support shield assembly
- Core barrel assembly
- Lower internals assembly

Based on the attributes of the 171 items evaluated in the FMECA tables, 26 items fall into moderate and significant risk bands (III and IV) based on safety consideration, and 71 fall into bands from moderate to extreme (III to V) based on economic consideration. These results show that the majority of the internals component items are of low and insignificant safety and economic impact.

EPRI Perspective

The EPRI Materials Reliability Program Reactor Internals Focus Group (MRP RI-FG) has been conducting studies to develop technical bases to support aging management of PWR internals (of B&W, Westinghouse, and CE designs), with attention to utility license renewal commitments. This component item FMECA document is one of a series of reports to provide a basis for developing PWR internals inspection and evaluation (I&E) guidelines for utility applications. The results documented here provided a basis for categorization and ranking results described in MRP-189, *Materials Reliability Program: Screening, Categorization, and Ranking of B&W-Designed PWR Internals Component* (EPRI report 1013232, September 2006). Other related EPRI reports include the following: *Materials Reliability Program: Screening, Categorization, and Ranking of Reactor Internals Components for Westinghouse and Combustion Engineering PWR Designs (MRP-191)* (EPRI report 1013234, September 2006); *Materials Reliability Program: Framework and Strategies for Managing Aging Effects in PWR Internals (MRP-134)* (EPRI report 1008203, June 2005); *Materials Reliability Program: Inspection and Flaw Evaluation Strategies for Managing Aging Effects in PWR Internals (MRP-153)* (EPRI report 1012082, December 2005); and *Materials Reliability Program: PWR Internals Material Aging Degradation Mechanism Screening and Threshold Values (MRP-175)* (EPRI report 1012081, December 2005).

Keywords

Materials Reliability Program
PWR Internals
B&W Design
Degradation Mechanism
Categorization and Ranking
Failure Modes, Effects, and Criticality Analysis (FMECA)
Risk

ABSTRACT

The purpose of this report is to develop and present the results of a failure modes, effects, and criticality analysis (FMECA) of Babcock & Wilcox (B&W)-designed pressurized water reactor (PWR) internals. While the FMECA treats the B&W plants on a generic basis, plant-specific information, when readily known, was included. The results are pairs of internals component items and age-related degradation mechanisms organized into risk bands to identify the internals component items subject to a specific age-related degradation mechanism (ARDM) that significantly contributes to either safety or economic risk of a B&W-designed nuclear power plant. The results from the FMECA will be used to help provide the technical bases for screening, ranking, and categorization for age-related degradation mechanisms of PWR internals items.

The B&W-designed PWR internals consists of two major structural assemblies that are located within, but not integrally attached to (i.e., not welded to) the reactor vessel. These major assemblies are the plenum assembly and the core support assembly (CSA). For discussion purposes, the CSA is presented as three principle sub-assemblies: the core support shield (CSS) assembly, the core barrel assembly, and the lower internals assembly. Each of these assemblies is discussed in greater detail.

The objective of this analysis is to provide a systematic review of the B&W-designed PWR internals to identify combinations of internals component items and age-related degradation mechanisms that potentially result in degradation leading to significant risk. A FMECA approach was used in which inductive reasoning ensures that the effects of all component items and their failure modes are examined. An appropriate level of detail is selected, and all "component items" at that level of detail are enumerated to produce a mutually exclusive and complete rendering of the entire "system" under study. For each component item, a complete set of failure modes is specified, and the effect(s) of each failure mode on the system is determined. From this, each failure mode can be judged on its importance to risk, based on the susceptibility (likelihood of the degradation mechanism) and severity of consequences. For this FMECA, consequences were examined from two perspectives: safety and economic. An expert panel was used to assign the semi-quantitative susceptibility and consequence metrics. Common cause failures and cascading (dependent) failures were also considered. Results are summarized by enumerating the number of component item/ARDMs pairs that are in each risk band for safety and economic consequences. The results are generally consistent with the previous IMT analysis and with the level of redundancy evident in the PWR internals design.

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- Dick Deveney, Supervisory Engineer — Neutronics/Safety
- Steve Fyfitch, Advisory Engineer — Materials
- Bill Gray, Project Manager — Project Management
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- Ken Moore, Advisory Engineer — Materials
- Wayne Pavinich, P.E., Advisory Engineer — Materials
- Eric Polstra, Principal Engineer — Safety Analysis
- Mark Rinckel, Project Manager — License Renewal
- Jeffery Seals, Advisory Engineer — Safety/Accident Analysis
- Sudhir Shah, Ph.D., Advisory Engineer — Stress, Dynamics
- Jim Smotrel, Advisory Engineer — Thermal-Hydraulics
- J. R. (Tuck) Worsham, Advisory Scientist — Radiation Physics
- Hongqing Xu, Engineer — Materials

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ACRONYMS AND ABBREVIATIONS

Acronym/Abbreviation	Definition
ANO-1	Arkansas Nuclear One Unit 1
ARDM	Age-Related Degradation Mechanism
B&W	Babcock & Wilcox
CCF	Common Cause Failure
CR-3	Crystal River Unit 3
CRA	Control Rod Assembly
CRGT	Control Rod Guide Tube
CSA	Core Support Assembly
CSS	Core Support Shield
DB	Davis-Besse
DM	Degradation Matrix
EPRI	Electric Power Research Institute
FMECA	Failure Modes, Effects, and Criticality Analysis
HAZ	Heat-Affected Zone
IASCC	Irradiation-Assisted Stress Corrosion Cracking
ID	Inside Diameter
IE	Irradiation Embrittlement
IMI	Incore Monitoring Instrumentation
IMT	Issue Management Table
IP	Issue Programs
LOCA	Loss of Coolant Accident
MDA	Materials Degradation Assessment
MEOG	Material Executive Oversight Group
MRP	Materials Reliability Program
MTAG	Materials Technology Advisory Group

OCL	Operational Cyclic Loading
OD	Outside Diameter
ONS	Oconee Nuclear Station (Units 1, 2, and 3)
PWR	Pressurized Water Reactor
RCS	Reactor Coolant System
RI-FG	Reactor Internals Focus Group
RiT	Reduction in Toughness
RV	Reactor Vessel
SCC	Stress Corrosion Cracking
SR/IC	Irradiation-Enhanced Stress Relaxation and Creep
SSHT	Surveillance Specimen Holder Tube
T&ISR/C	Thermal & Irradiation-Enhanced Stress Relaxation and Creep
TE	Thermal Aging Embrittlement
TSR/C	Thermal Stress Relaxation and Creep
TMI-1	Three Mile Island Unit 1
VS	Void Swelling

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1 INTRODUCTION

1.1 Report Purpose

The purpose of this report is to develop and present the results of a failure modes, effects, and criticality analysis (FMECA) of Babcock & Wilcox (B&W)-designed pressurized water reactor (PWR) internals. This is a precursor document to the screening and categorization process in reference [9]. The plants considered in this project are: Arkansas Nuclear One Unit 1 (ANO-1), Crystal River Unit 3 (CR-3), Davis-Besse (DB), Oconee Nuclear Station Units 1, 2, and 3 (ONS), and Three Mile Island Unit 1 (TMI-1). While the FMECA treats the B&W-design plants on a generic basis, plant-specific information, when readily known, was included. The results are pairs of internals component items and age-related degradation mechanisms organized into risk bands to identify the internals component items subject to a specific age-related degradation mechanism (ARDM) that significantly contributes to either safety or economic risk of a B&W-designed nuclear power plant. The results from the FMECA will be used to help provide the technical bases for screening, ranking, and categorization for age-related degradation mechanisms of PWR internals items. This report was prepared under the direction and sponsorship of the Electric Power Research Institute (EPRI) Materials Reliability Program (MRP) Reactor Internals Focus Group (RI-FG).

This report is one element in an overall strategy for managing the effects of aging in PWR internals using knowledge of internals design, materials and material properties, and applying screening methodologies for known aging degradation mechanisms. Related MRP documents include a Framework and Strategy for Managing Aging Effects in PWR Internals [1], Inspection and Flaw Evaluation Strategies for Managing Aging Effects in PWR Internals [2], and PWR Internals Material Degradation Mechanism Screening and Threshold Values [3].

1.2 Background

NEI 03-08 [4] is a materials management guideline that became effective on January 2, 2004. This document outlines the policy and practices that the industry has committed to follow in managing materials' aging issues. Two standing committees were established to assist the utilities and the issue programs (IPs) they fund. The Materials Technology Advisory Group (MTAG) provides technical oversight and the Materials Executive Oversight Group (MEOG) provides executive oversight. Neither of these groups is directly involved in the technical work, which resides in the IPs.

Recently, an industry ad hoc committee was tasked by the MTAG to prepare a generic degradation matrix (DM) applicable to all PWR internals designs [5]. Expert elicitation, laboratory studies, and field experience were used to identify potential mechanisms by which each of the PWR internals materials, among other materials and component items, might degrade. The current DM groups the age-related degradation mechanisms into several broad categories such as stress corrosion cracking (SCC), corrosion and wear, fatigue, and reduction in toughness (RiT). Each of these is comprised of various subcategories of degradation mechanisms. For example, the RiT category includes thermal aging embrittlement and void swelling.

The currently identified age-related degradation mechanisms considered in the FMECA, also considered for component items screening [3], are as follows: [1]

- Stress corrosion cracking (SCC)
- Irradiation-assisted SCC (IASCC)
- Wear
- Fatigue
- Thermal aging embrittlement (TE)
- Irradiation embrittlement (IE)
- Void swelling (VS)
- Irradiation-enhanced stress relaxation and creep (SR/IC)

The DM was used as input to a Materials Degradation Assessment / Issue Management Table (MDA/IMT) ad hoc committee. This committee developed the IMTs for reactor coolant system components, including a PWR internals IMT [6]. The FMECA extends the insights gained from the development of the IMT; by considering the spectrum of values related to susceptibility and consequence (severity), and a risk matrix was developed that permitted additional ranking capability. It should be noted that while the IMT consequences were looking at specific events occurring (A to G) [2], it would be hard to justify any ranking with the IMT results, as there were no levels of "degrees" in either the susceptibility or the consequences. The FMECA and the resulting risk matrix provide the context in which to discriminate between different pairs of age-related degradation mechanism/component item, whereas the IMT effort was not designed to do that. Chapter 4 and Appendix B provide more discussions on the IMT and FMECA, and their relationships.

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[1] The more generally known acronyms provided in this list will be used throughout the remainder of this report, in lieu of the acronyms defined in the industry DM.

[2] In IMT [6, 8], Adverse Consequences of Failure are summarized using letters A to G defined as follows:

(A) Precludes the ability to reach safe shutdown
(B) Causes a design basis accident
(C) Causes significant onsite and/or offsite exposure
(D) Jeopardizes personnel safety
(E) Breaches reactor coolant pressure boundary
(F) Breaches fuel cladding
(G) Causes a significant economic impact

These categorizations can filter out component items that are of no consequences but they do not provide a sufficient basis of ranking.

The results of the FMECA will be an element in the overall screening criteria that will be used to categorize all PWR internals component items in accordance with the strategy developed in MRP-134 (Figure 4-1) [1]. In the categorization process, analysts will develop a mapping between the risk matrix bands, and perhaps even specific risk matrix cells, and the initial three component item categories, e.g., Categories, A, B, and C. This mapping will be based on the original screening results and the definitions of the three screening categories.

1.3 Report Structure

Chapter 2, which is adapted from MRP-157 [8] to facilitate discussion, provides an overall description of the B&W-designed PWR internals. The description is divided into four major groups, as shown in Figure 1-1. The plenum assembly and the lower internals assembly are further divided into sub-assemblies. Chapter 3 provides the analytical approach used in the development of the FMECA, provides a description of each table header, and the development of the risk matrix. Chapter 4 lists the key assumptions used to populate the FMECA table. A summary of the FMECA results is provided in Chapter 5. Appendix A contains the entire FMECA, as four separate tables. Appendix B provides a table that highlights the specific differences between the IMT approach and the FMECA.

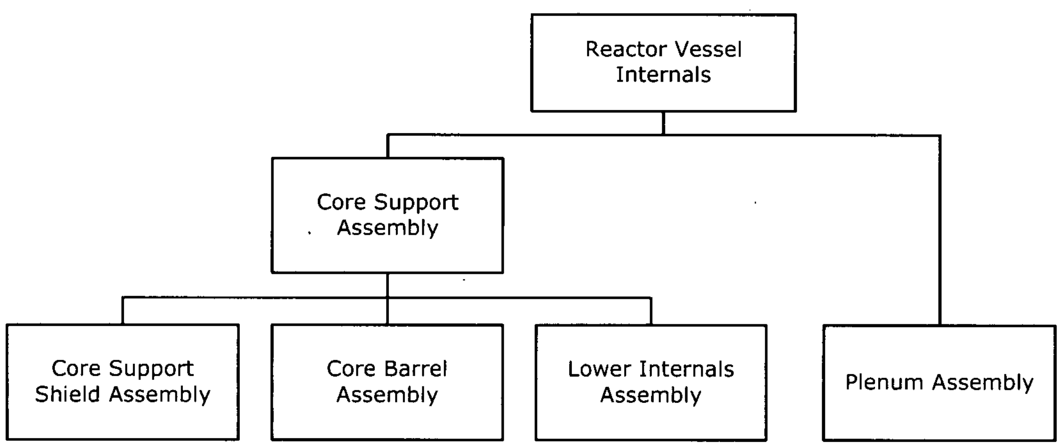

Figure 1-1
B&W-Designed PWR Internals Assemblies

2 GENERAL DESCRIPTION OF B&W-DESIGNED PWR INTERNALS

The B&W-designed PWR internals consist of two major structural assemblies that are located within, but not integrally attached to (i.e., not welded to) the reactor vessel. These major assemblies are the plenum assembly and the core support assembly (CSA). For discussion purposes, the CSA is presented as three principle sub-assemblies: the core support shield (CSS) assembly, the core barrel assembly, and the lower internals assembly. The general arrangement of the B&W-designed PWR internals is presented in Figure 2-1. The description of the internals in this section is taken directly from Reference [8].

2.1 Scope and General Discussion

The plenum assembly is a cylindrical assembly with perforated grids on top and bottom. The plenum assembly fits inside the CSS, positions the fuel assemblies, and provides the core hold-down required for hydraulic lift forces. The plenum assembly provides continuous guidance and protection of the control rods. In addition, the plenum assembly directs flow out of the core to the reactor vessel (RV) outlet nozzles. The plenum assembly is removed every refueling outage to permit access to the fuel assemblies.

The CSA remains in place in the reactor vessel and is only removed to perform scheduled inspections of the RV interior surfaces or of the CSA. The CSA is assembled from three separate sub-assemblies, which bolt together to form one tall cylinder.

The CSS assembly is the top portion of the CSA. It is a cylinder with an upper flange that rests on a circumferential support ledge in the RV closure flange and supports the entire CSA. The core barrel assembly is a second cylinder bolted to the bottom of the CSS assembly. The 177 fuel assemblies that make up the core are loaded into the core barrel assembly. The lower internals assembly is bolted to the bottom of the core barrel assembly. The lower internals support the core and direct the coolant flow up past the fuel assemblies. In addition, the lower internals provide guidance of the incore monitoring instrumentation from the reactor vessel interface to the lower fuel assembly end fitting.

The PWR internals assemblies discussed above and the bolting joining the sub-assemblies are within the scope of this project. The welds considered in this project include major structural welds that form or join the major cylinders and flanges, and minor structural welds joining parts such as lifting lugs, support pipes and tubes to the major sub-assemblies. There are no pressure-retaining or pressure boundary welds associated with the PWR internals.

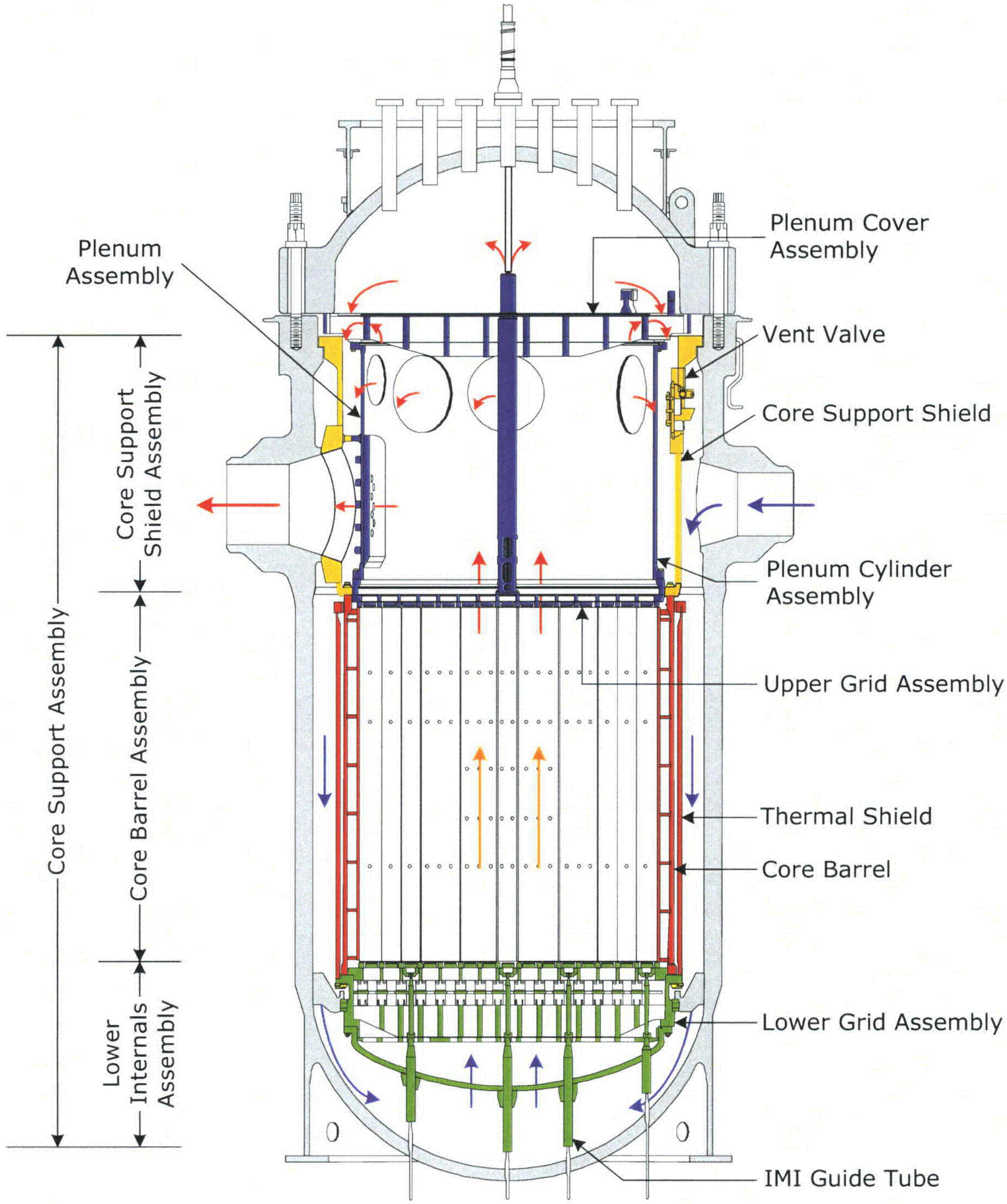

Figure 2-1
B&W-Designed PWR Internals General Arrangement

(Note: some component items are rotated for clarity)

2.2 Component Item Function

The PWR internals serve a number of functions. The CSA provides physical support and orientation for the reactor core (fuel assemblies), the control rod assemblies, and the incore monitoring instrumentation. All static and dynamic loads from the assembled component items and fuel assemblies are carried by the CSA and transferred to the reactor vessel closure flange. It also acts as a flow boundary to direct incoming RCS coolant from the cold leg inlet nozzles down the annulus formed between the CSA and the inner RV wall to the lower plenum below the CSA. Once inside the CSA, it guides the coolant up through the core, into the upper plenum region above the core, and through the outlet nozzles to the hot leg piping (see Figure 2-1). Finally, the CSA provides neutron and gamma shielding for the reactor vessel. CR-3 and DB also have surveillance specimen holder tubes (SSHTs) that provide positioning and support for the reactor vessel irradiation specimens.

2.3 Plenum Assembly

The plenum assembly is a cylindrical assembly approximately 11 feet tall, located inside the CSS and directly above the reactor core. This assembly holds down and aligns the fuel assemblies, directs the flow of reactor coolant from the core to the reactor vessel outlet nozzles, and supports the 69 control rod guide tube (CRGT) assemblies. It is made up of the following assemblies: the plenum cover assembly, the plenum cylinder assembly, the upper grid assembly, and the CRGT assemblies, as shown in Figure 2-2. The plenum assembly must be removed in order to access the fuel assemblies.

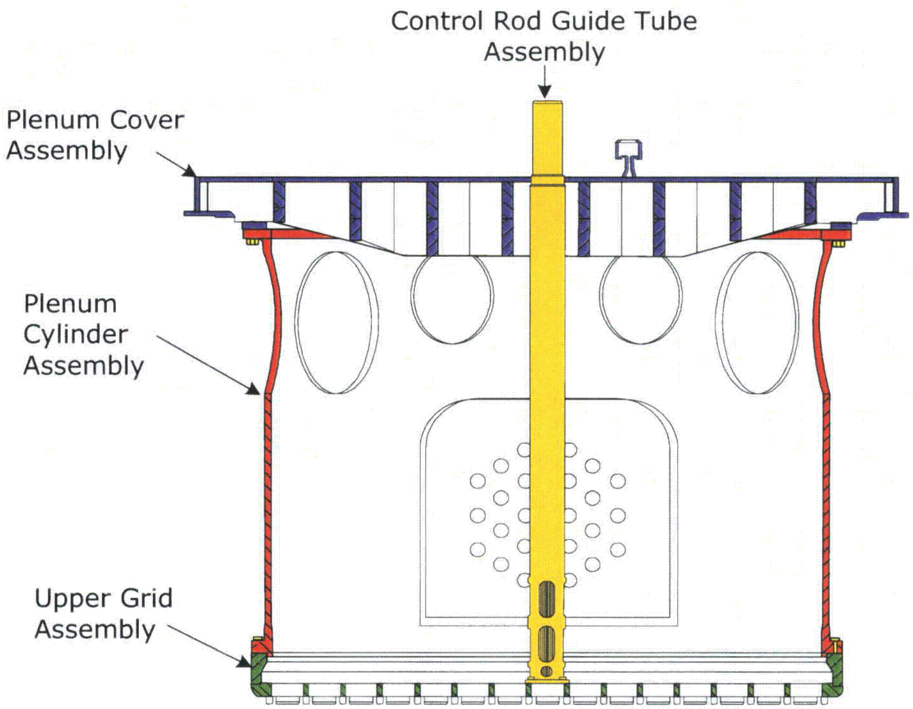

Figure 2-2
Plenum Assembly

2.3.1 Plenum Cover Assembly

The plenum cover assembly is bolted to the top of the plenum cylinder. It consists of a weldment, a bottom flange, a support ring and flange, a cover plate, and lifting lugs. It provides support for the top of the 69 CRGT assemblies. The lifting lugs are used to lift the plenum assembly out of the reactor vessel. Figure 2-3 shows the plenum cover assembly and each component item is described below.

The plenum cover weldment is a lattice assembled from two sets of ten parallel flat plates intersecting perpendicularly with ten-inch spacing between ribs. The individual ribs are two-inch thick flat stainless steel plates of varying lengths and heights. Rib (compression) pads are welded to the top outer edge of each rib, forming a mating surface for the reactor vessel head.

The plenum cover support flange is welded to the bottom of the plenum cover weldment assembly. It provides the seating surface that rests on top of the CSS assembly and against the inner RV wall. At each of the four axis locations, the support flange has keyways that mate with reactor vessel flange keys to align the plenum assembly with the reactor vessel, the reactor closure head control rod drive penetrations, and the CSA.

The plenum cover support ring is a two-inch thick ring, welded onto the top of the support flange and outer vertical edges of the plenum cover weldment. The support ring provides a surface that mates with the reactor vessel head. At each of the four axis locations, the support ring has keyways that mate with closure head key blocks to align the closure head assembly.

The plenum cover bottom flange is a flat ring welded to the bottom of the weldment to provide a surface to attach the plenum cylinder. It is located inside of the plenum cover support flange, and has 64 tapped holes to which the upper flange of the plenum cylinder is bolted.

The plenum cover plate is a ½-inch thick disk that is welded to the top center of the plenum cover weldment. It has 69 holes through which the tops of the CRGTs are fitted and welded. The cover plate size allows some reactor coolant flow up past the plenum cover into the upper reactor head region.

Three lifting lugs are spaced 120° apart around the top of the plenum cover assembly, and are used to remove the plenum assembly. There are two types of lifting lug arrangements. In all plants but Oconee Nuclear Station-1 (ONS-1), T-shaped lifting lugs are fastened to base blocks with two bolts that are secured with locking cups. The base blocks are welded between two of the weldment ribs. At ONS-1, each lifting lug is a single piece, which is similarly welded between ribs on the plenum cover weldment.

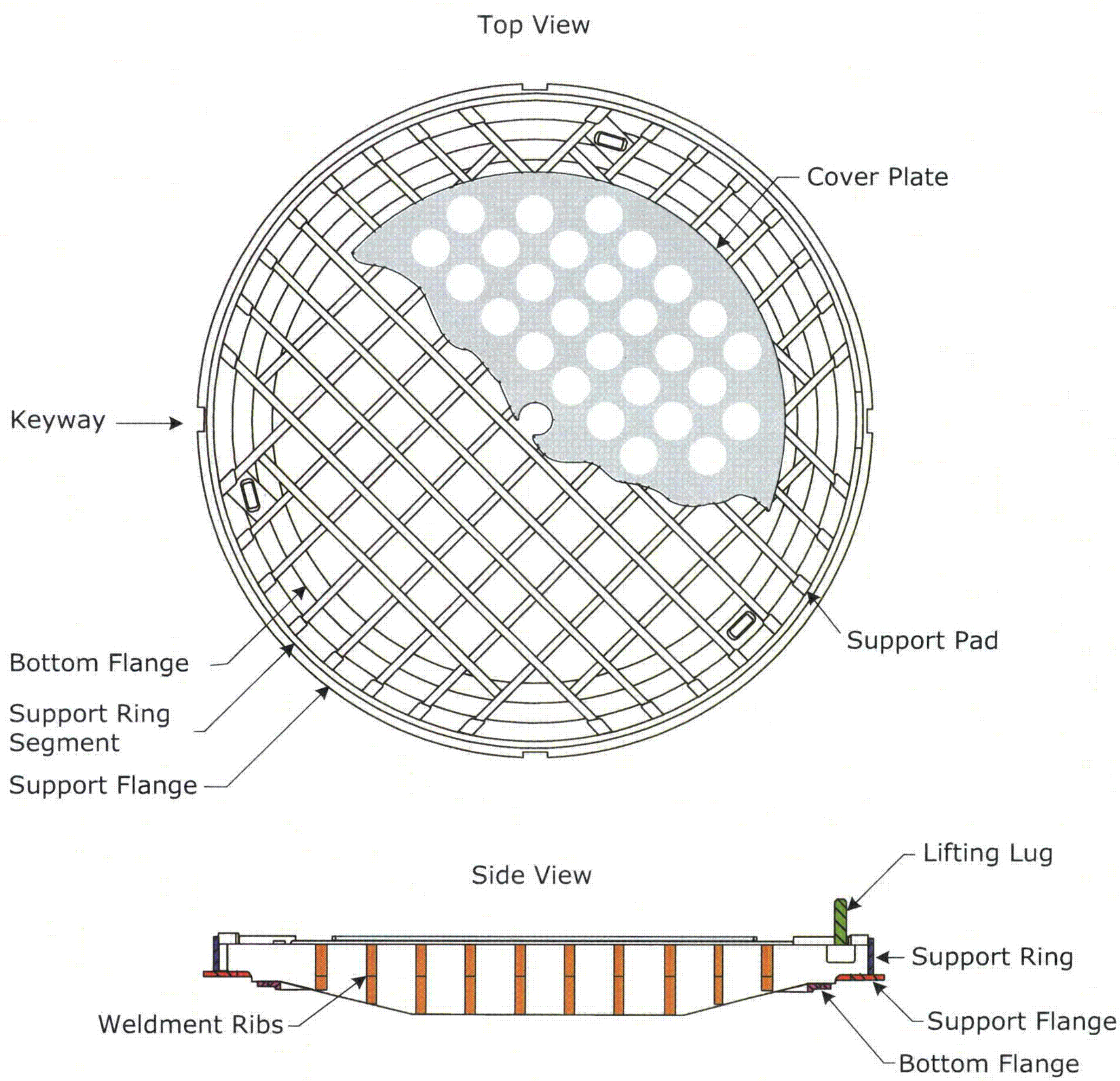

Figure 2-3
Plenum Cover Assembly

2.3.2 Plenum Cylinder Assembly

The plenum cylinder assembly is bolted to the bottom of the plenum cover assembly and consists of a cylinder, top and bottom flanges, reinforcing plates, and round bars. It directs the flow of reactor coolant from the core area to the reactor vessel outlet nozzles. The plenum cylinder assembly is shown in Figure 2-4, and each component item is described below.

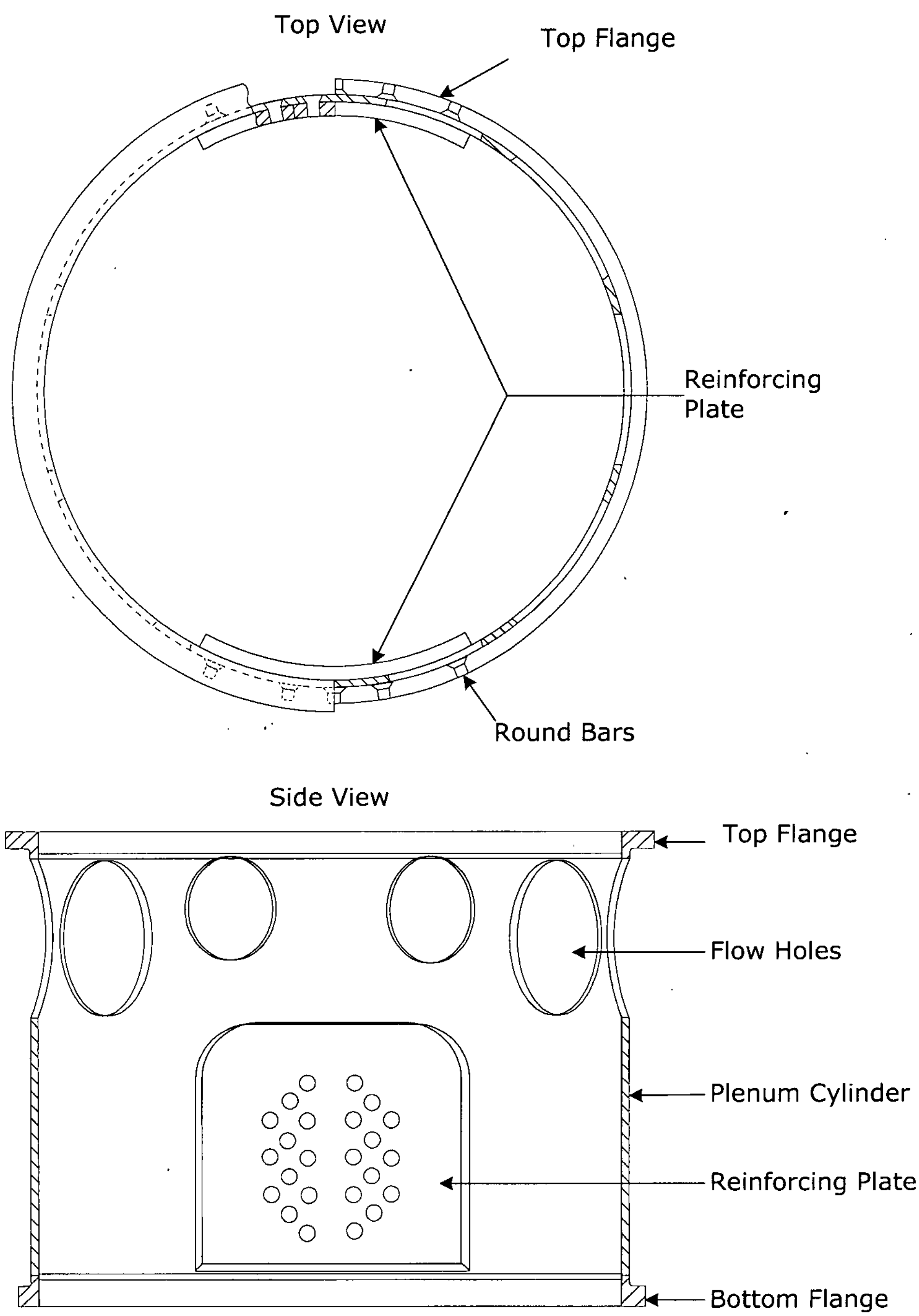

Figure 2-4
Plenum Cylinder Assembly

The plenum cylinder is fabricated from 1½-inch thick stainless steel plate. The plenum cylinder has 24 small holes at each of two locations to permit some of the reactor coolant coming up into the plenum to flow directly to the outlet nozzles. The majority of the reactor coolant passes through ten large holes (six 34 inches in diameter and four 22 inches in diameter) at the top of the cylinder, out into the annulus between the plenum cylinder and the CSS, and ultimately down and out through the reactor vessel outlet nozzles.

The plenum cylinder top flange is welded to the top of the plenum cylinder. The plenum cover assembly is bolted to the top flange with 64 bolts held in place with locking cups.

The plenum cylinder bottom flange is welded to the bottom of the plenum cylinder. The plenum upper grid assembly is bolted to the bottom flange with 36 bolts held in place with locking cups.

Two three-inch thick reinforcing plates are welded to the inner surface of the plenum cylinder. They are aligned with reactor vessel outlet nozzles and have 24 holes aligned with the plenum cylinder holes described previously. The reinforcing plates and holes are installed to help the structure withstand the blowdown loads associated with a hot leg large break loss of coolant accident (LOCA).

To ensure that the flow path between the plenum cylinder and the CSS is maintained during a transient, a set of 13 small stainless steel round bars or lugs are welded to the outer surface of the plenum cylinder at each of the outlet nozzle areas. These lugs are positioned opposite similar lugs welded to the inner surface of the CSS. The round bars are four inches long and 2½ inches in diameter and are frequently referred to as "LOCA lugs" or "LOCA bosses."

2.3.3 Upper Grid Assembly

The upper grid assembly sits inside the lower flange of the CSS and is bolted to the plenum cylinder bottom flange. It provides the support and seating surface for the tops of the fuel assemblies located in the core barrel below, and provides restraint and alignment for the bottoms of the CRGT assemblies. It consists of an upper grid ring forging, an upper grid rib section, and fuel assembly support pads, as shown in Figure 2-5. Each component item is described below.

The upper grid ring forging is a ring with an inward flange on the upper end. The top of the upper grid ring forging is machined to accept the 36 bolts fastening the upper grid assembly to the plenum cylinder bottom flange, described previously. The upper grid rib section is fastened to the bottom of the upper grid ring forging with 36 cap screws held in place by welded locking pins.

The upper grid rib section is a three-inch thick disk with 177 squares machined out, leaving a grid of one-inch wide "ribs." The square holes align with the fuel assembly locations in the core below. Pads to support and align the fuel assemblies are doweled and bolted into the ribs on the bottom side. The topside of the rib section is drilled and tapped to accept the dowels and cap screws, which hold the bottom flange of the 69 CRGT assemblies to the upper grid.

There are 384 fuel assembly support pads attached to the bottom of the upper grid rib section to provide a seating surface and support for the tops of the fuel assemblies. The pads are each held in place by two dowels and a cap screw, which are subsequently welded in place.

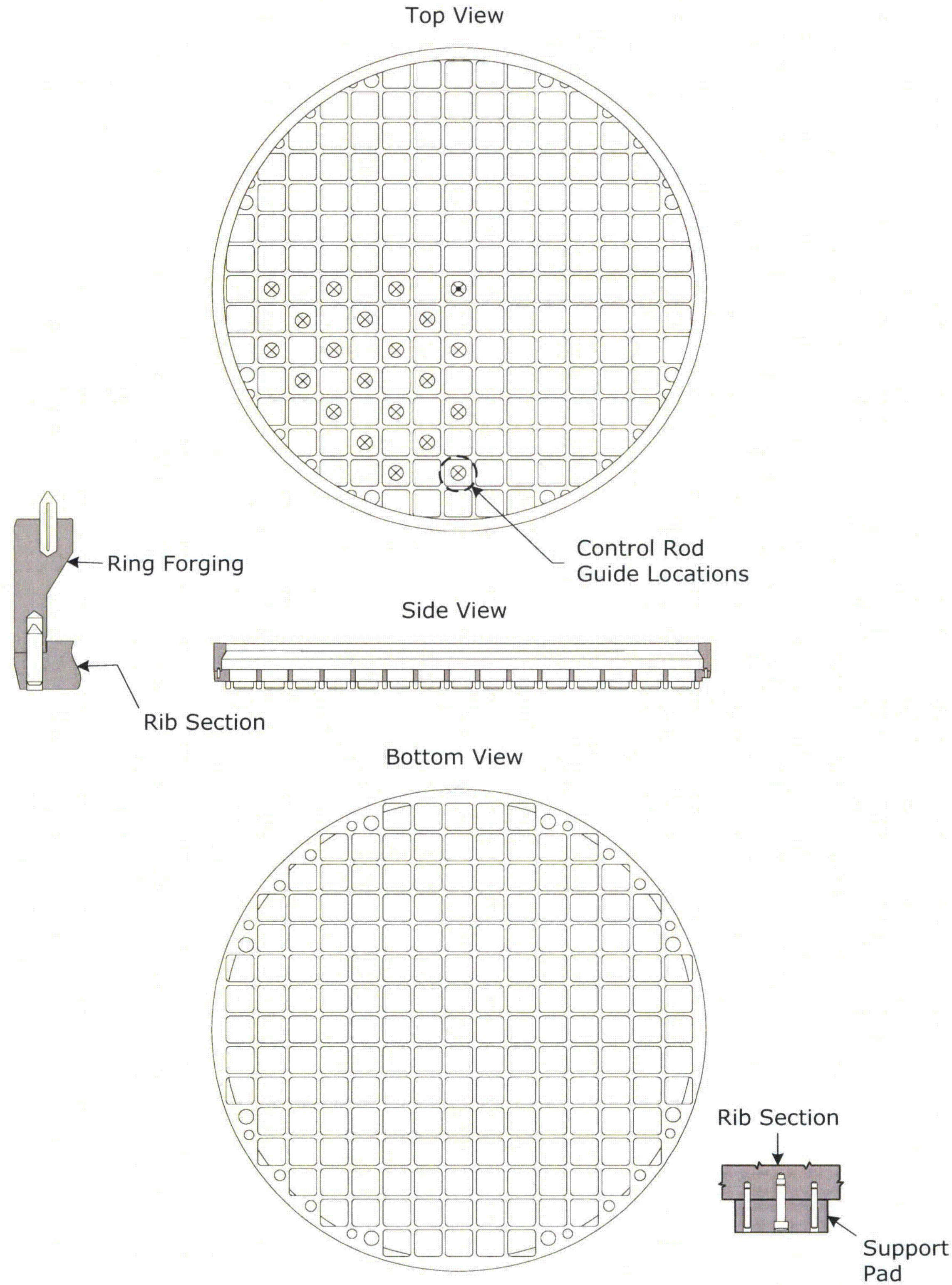

Figure 2-5
Upper Grid Assembly

2.3.4 Control Rod Guide Tube Assembly

The 69 vertical CRGT assemblies are welded to the plenum cover plate and bolted to the upper grid. It consists of a pipe (or guide housing), a flange, spacer castings, guide tubes, and rod guide sectors. The CRGT assemblies provide control rod assembly (CRA) guidance, protect the CRA from the effects of coolant cross-flow, and structurally connect the upper grid assembly to the plenum cover. Design clearances in the guide tube accommodate misalignment between the guide tubes and the fuel assemblies.

The top end of each of the 69 CRAs consists of a spider plate, through which 16 individual control rods are suspended. As shown in Figure 2-6, the 139-inch long control rods are arranged in two concentric rings, four rods in the middle ring and 12 in the outer ring. The rods have no other vertical support other than the spider plate at the top. (The control rod assemblies and the control rod drive mechanisms are not within the scope of this project.) The CRGT assembly provides support both for the CRA as a whole and for each of the 16 individual control rods within each CRA.

The outer portion of the CRGT assemblies consists of pipes (or guide housings) welded to the CRGT assembly flanges at the bottom ends. The inside of each assembly consists of an internal sub-assembly with ten parallel horizontal spacer castings to which are brazed 12 perforated vertical rod guide tubes and four pairs of vertical rod guide sectors, also called "C-tubes." These internal sub-assemblies of spacers, rod guide tubes and rod guide sectors are referred to as the "control rod guide brazements." Figure 2-7 shows the CRGT assembly spacer castings and the control rod guide brazement configuration.

The CRGT assembly pipes (or guide housings) are approximately 12 feet long, eight-inch diameter, stainless steel. At ten elevations, they are drilled at four equally spaced circumferential locations to accommodate the cap screws that hold the spacer castings in place.

Four equally spaced three-inch diameter holes are located two inches above the bottom of the CRGT assembly pipes. Above them are two rows of four three-inch wide, 8¾-inch high oval-shaped holes. These holes allow some of the reactor coolant traveling up the pipes to exit out into the plenum and to ensure that the pressures are equalized on both sides of the pipes and prevent hydraulic effects from impeding control rod travel.

The pipes are welded to the top of the plenum cover plate. The top of the pipes extend approximately 21 inches above the plenum cover plate into the upper head area. The CRGT assembly pipes are shown in Figure 2-1 and Figure 2-8.

The CRGT assembly flanges are 1¼-inch thick square plates with a hole in the center to match the inner diameter of the CRGT assembly pipes. Four additional small semicircular flow paths are equally spaced about the center to permit reactor coolant system (RCS) flow upward through the flange on the outside of the CRGT assembly pipe. Each flange is drilled to accept two dowels and four hex head cap screws for attachment to the upper grid rib section.

The CRGT assembly spacer castings are ¾-inch thick disks, with internal spaces to conform to the general shape of the control rod spider, with clearances to permit RCS flow and to accommodate the rod guide tubes and rod guide sectors.

Within each CRGT assembly are 12 rod guide tubes. These are long 0.750-inch inside diameter (ID), 0.095-inch thick tubes with a 0.3125-inch wide vertical slot. The tubes have a vertical row of 99 ¼-inch holes spaced at ½-inch increments (from the bottom of the tube) to permit RCS flow into the area to balance the pressure on the inside and outside of the tube to prevent the control rod from being pulled into the tube's slot due to a differential pressure. The rod guide tubes are brazed into holes in the spacer castings, with the slots aligned to match where the spider arms pass.

The CRGT assembly rod guide sectors are similar to the rod guide tubes, but are fabricated from 0.109-inch thick plates with a curved cross section. They are for the four inner individual control rods in each assembly that are suspended from the middle of a spider arm. They are brazed in pairs in holes in the spacer castings, facing each other with a gap between them to permit travel of the spider arm between them. The rod guide sectors do not have cooling holes like the rod guide tubes, since they are open on two sides.

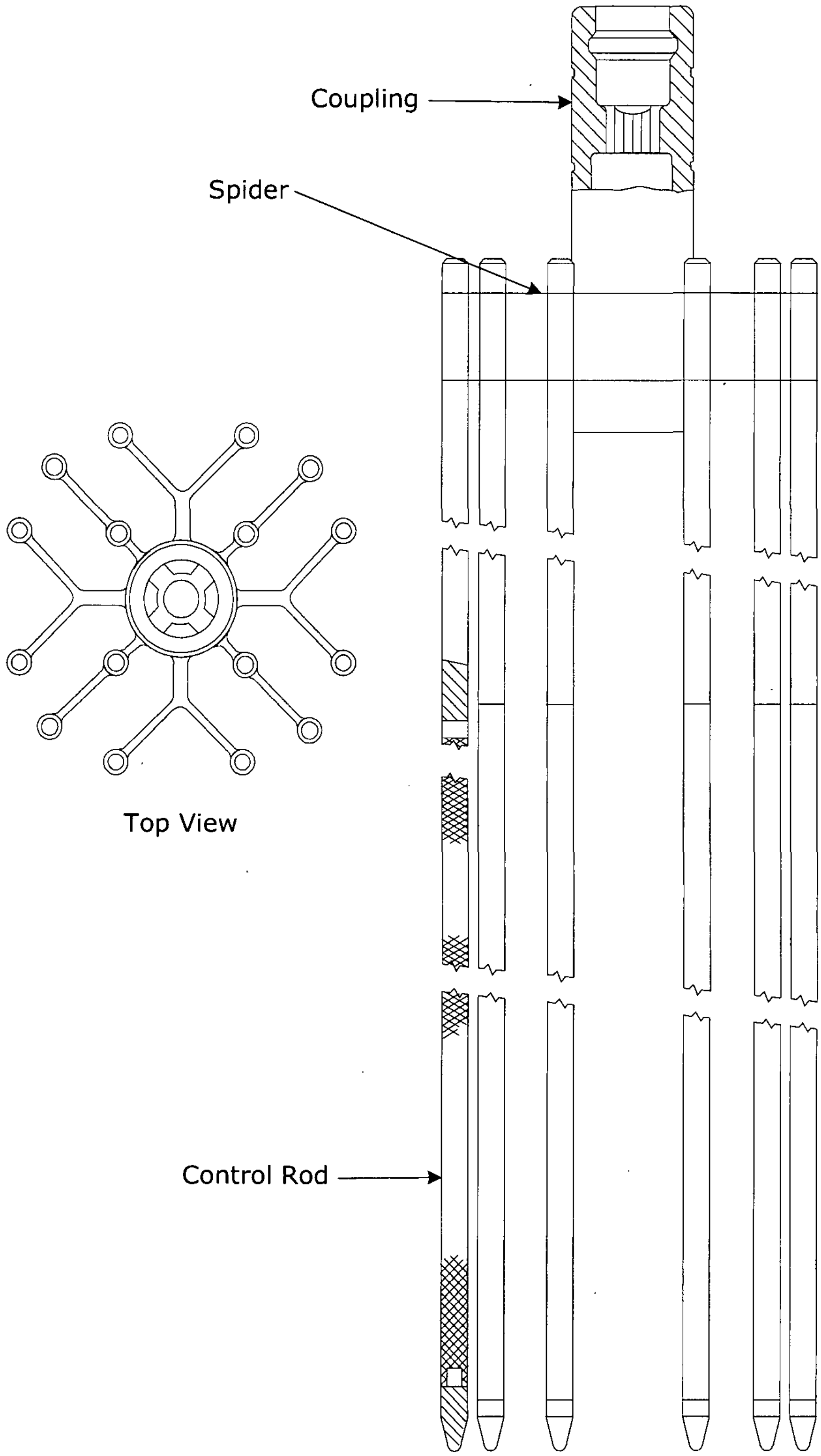

Figure 2-6
Control Rod Assembly

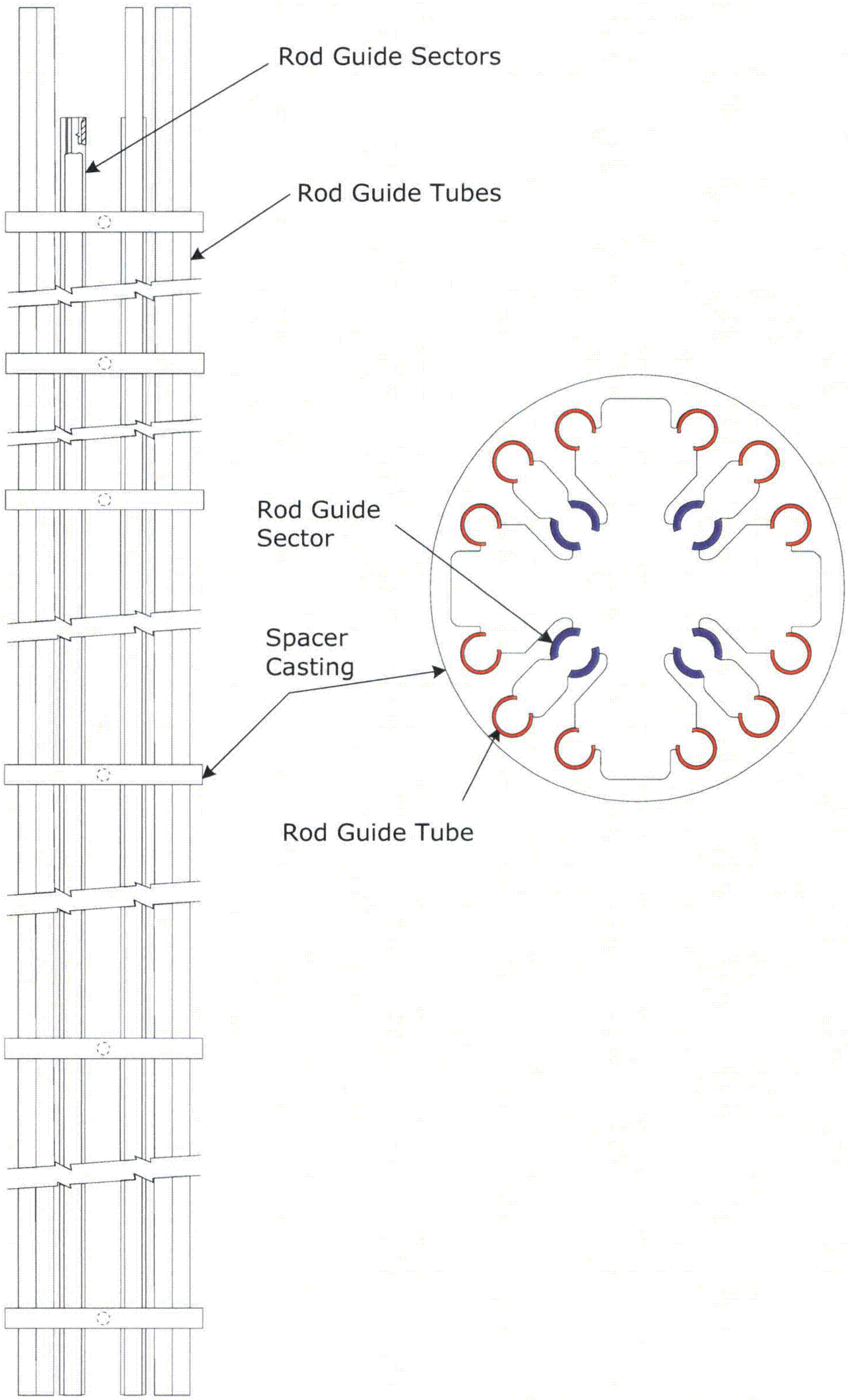

Figure 2-7
Control Rod Guide Brazement and Spacer Castings

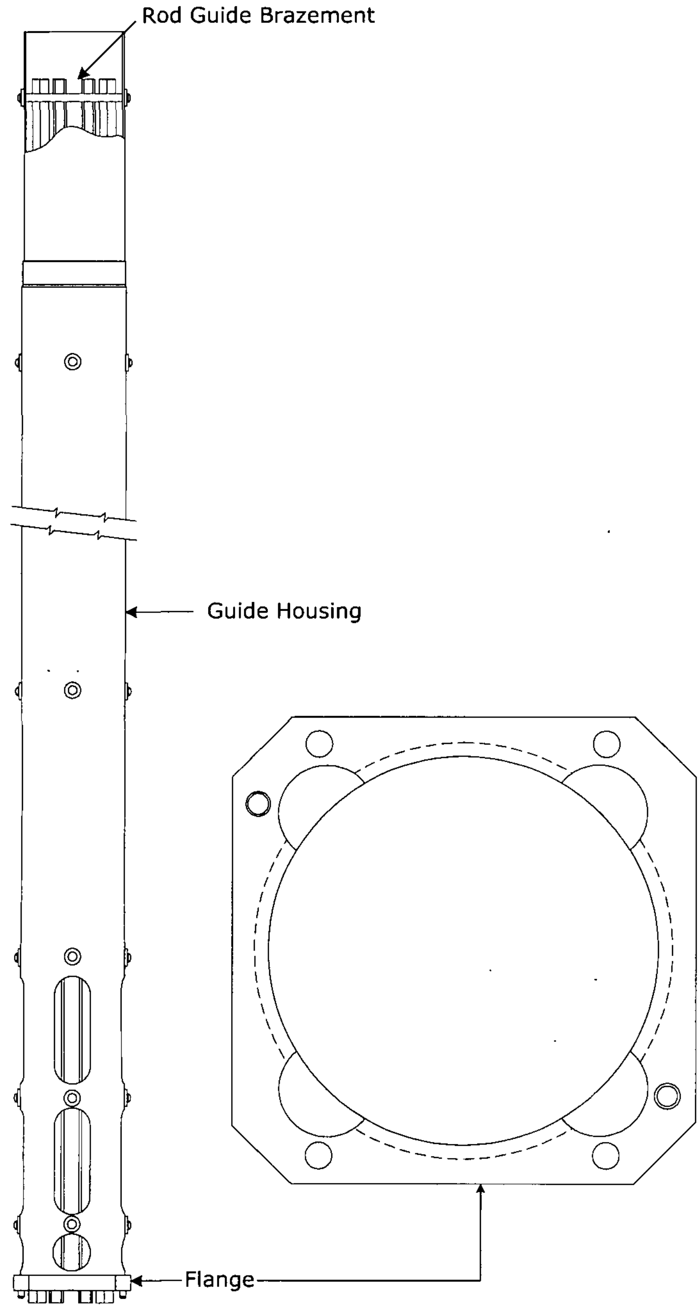

Figure 2-8
Control Rod Guide Tube Assembly

2.4 Core Support Shield Assembly

The CSS assembly is a flanged cylinder that sits on top of the core barrel. The CSS assembly provides a boundary between the incoming cold reactor coolant on the outside of the CSS assembly and the heated reactor coolant flowing on the inside of the CSS assembly (Figure 2-9). The CSS assembly consists of a cylinder, top and bottom flanges, outlet nozzles, vent valve nozzles, vent valves, round bars, flow deflectors, and lifting lugs.

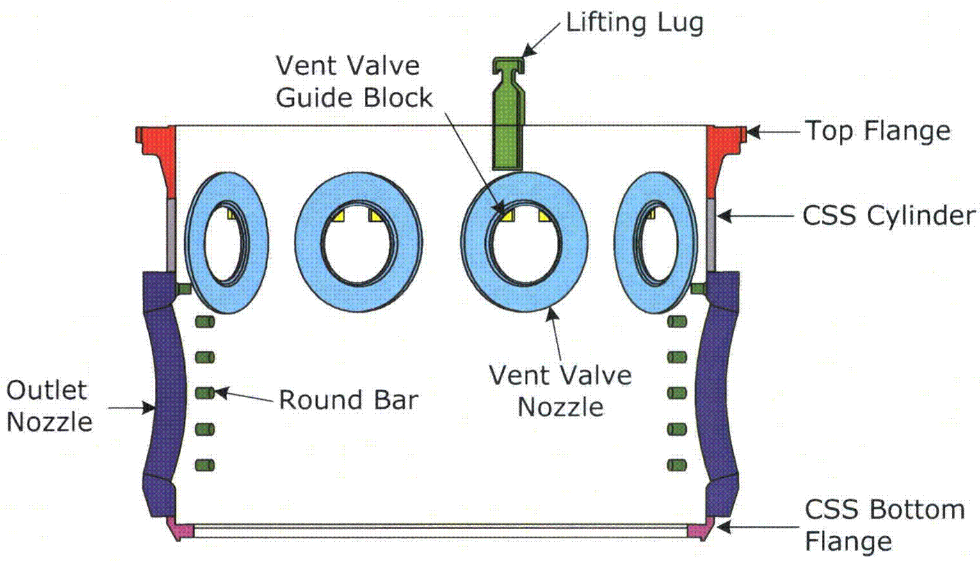

Figure 2-9
Core Support Shield Assembly

The plenum assembly is supported by and fits inside the CSS. The bottom flange of the CSS is bolted to the core barrel. The inside surface of the CSS bottom flange provides the lower seating surface for the plenum assembly.

The CSS cylinder wall has two openings with nozzles for RCS outlet flow. These openings are formed by two forged rings (ONS-3 uses a single casting) that seal to the reactor vessel outlet nozzles by the differential thermal expansion between the stainless steel CSS and the low-alloy steel RV. The nozzle seal surfaces are finished and fitted to a predetermined cold gap providing clearance for CSA installation and removal. At operating temperature, the mating metal surfaces are in contact to make a seal without exceeding allowable stresses in either the RV or internals.

The CSS top flange is welded to the top of the CSS cylinder. The CSS top flange extends out from the inner diameter. The bottom of the top flange rests on a circumferential ledge in the reactor vessel closure flange. The top of the flange provides the seating surface to support the bottom of the plenum cover support flange, and thus supports the entire plenum assembly. The bottom of the top flange is penetrated by the vent valve nozzles.

The CSS bottom flange is welded to the bottom of the CSS cylinder and bolted to the top flange of the core barrel with 120 core barrel bolts, secured with locking clips or locking cups. The bottom of the plenum assembly is guided by the inside surface of the CSS bottom flange.

The two outlet nozzles are 67 inch outside diameter (OD), 8¾-inch thick curved ring-shaped inserts that are welded into the CSS cylinder with full-penetration welds (i.e., the inner surfaces are welded flush with the inner cylinder wall and extend out horizontally approximately seven inches towards the inner RV wall). The wall thickness of the nozzle tapers, with the inner hole having an oval shape. The outlet nozzles at ONS-3 are castings; all other plants have forged nozzles.

At each outlet nozzle area, 13 round bars are located on the inner surface of the CSS cylinder to mate with the similar lugs welded on the outer surface of the plenum cylinder. The round bars ensure that the radial clearance between the two cylinders is maintained so RCS flow is not disrupted under any conditions. (These are frequently referred to as the "LOCA lugs" or "LOCA bosses.")

Eight (four at DB) vent valve nozzles (or mounting rings) are welded in the CSS cylinder wall. The vent valve nozzles provide support for the vent valve sub-assemblies. The nozzles are welded into the CSS cylinder using full-penetration welds. The nozzles are approximately 38-inch OD and are 6¼ inches long. To accommodate the vent valves, the inner surfaces of the rings have lips and flanges. Two small guide blocks are welded to the top outside surface of each vent valve nozzle. The guide blocks are machined to provide a small triangular seating surface for the vent valve assemblies.

Vent valve assemblies are installed in the mounting rings as shown in Figure 2-10. For all normal operating conditions, the vent valve is closed. In the event of a rupture of the reactor vessel inlet pipe, the valve will open to vent steam generated in the core directly to the break, thus permitting the core to be flooded and adequately cooled after emergency core coolant has been supplied to the reactor vessel.

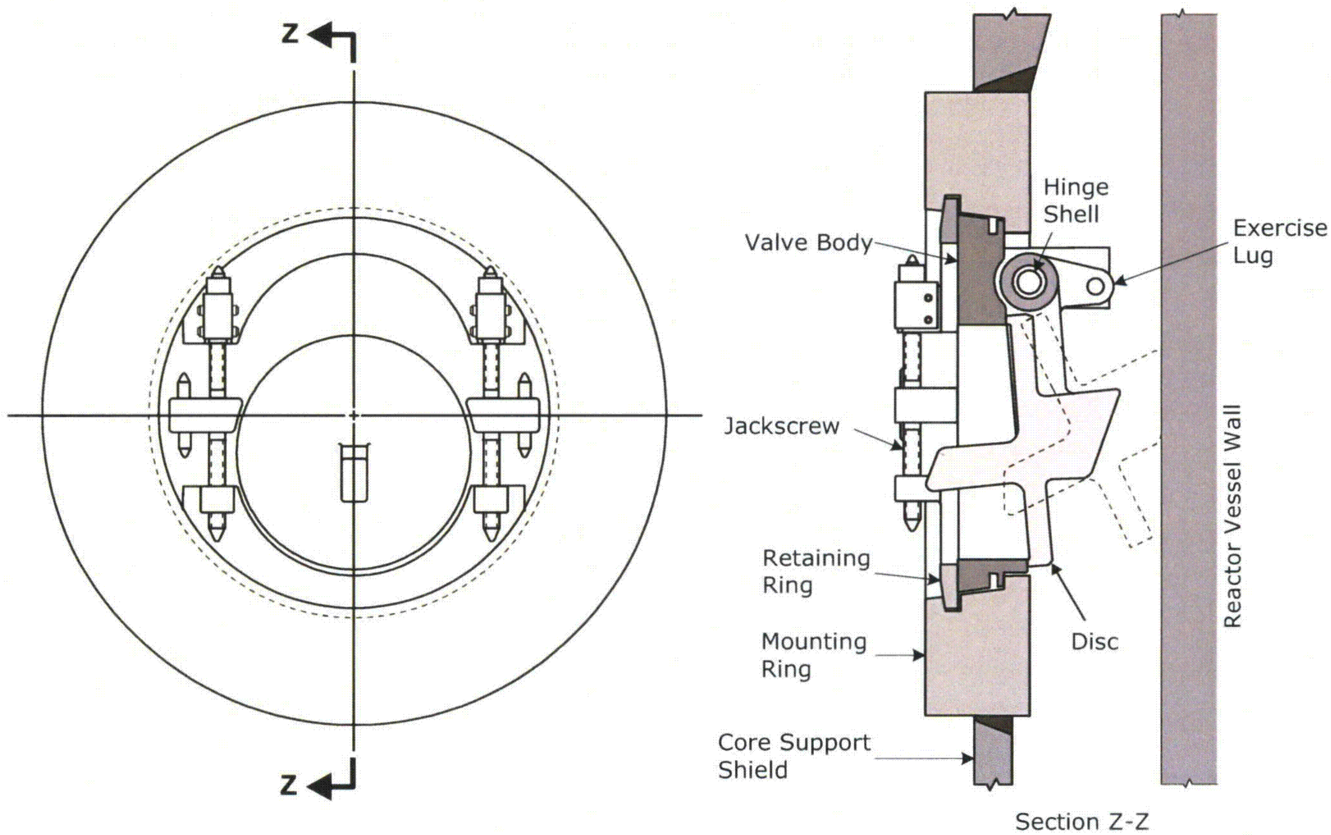

Figure 2-10
Vent Valve Assembly

Each vent valve assembly consists of a hinged disc, a valve body with sealing surfaces, a split-retaining ring and fasteners (which retain and seal the perimeter of the valve assembly), and an alignment device (to maintain the correct orientation). Each vent valve assembly can be remotely handled as a unit for removal or installation. Vent valve component parts, including the disc, are designed to minimize the possibility of loss of parts to the reactor coolant system, and all operating fasteners include a positive locking device. The hinged-disc includes a device for remote testing and verification of proper disc function. The external side of the disc is contoured to absorb the impact load of the disc on the RV inside wall without transmitting excessive impact loads to the hinge parts as a result of a LOCA.

The hinge assembly consists of a shaft, two valve body journal receptacles, two valve disc journal receptacles, and four flanged shaft journals (bushings). Loose clearances are used between the shaft and journal inside diameters, and between the journal outside diameters and their receptacles. The valve disc journal contains integral exercise lugs for remote operation of the disc with the valve installed in the CSS. The hinge assembly provides eight loose rotational clearances to minimize any possibility of impairment of free motion of the disc in service. In addition, the valve disc hinge loose clearances permit disc self-alignment so that the external differential pressure adjusts the disc seal face to the valve body seal face. This feature minimizes the possibility of increased leakage and pressure-induced deflection loadings on the hinge parts in service.

The vent valve materials were selected on the basis of their corrosion resistance, surface hardness, anti-galling characteristics, and compatibility with mating materials in the reactor coolant environment. The jackscrews, once installed, may need to be cut out to replace the vent valve assembly. As such, vent valve assemblies with modified locking devices were made available.

A flow deflector, consisting of three one-inch thick plates shaped to form an inverted "U," is welded to the outer surface of the CSS cylinder around the area opposite each of the four inlet (cold leg) nozzles. These flow deflectors help divert the incoming flow downward to the bottom of the core, and minimize the upward flow that might damage the internal vent valve assemblies. The flow deflector plates were originally a uniform four-inch width (i.e., extended out four inches from the cylinder) and blocked most of the annulus between the CSS cylinder and the RV shell. Following hot functional testing at ONS-1, however, the side flow deflector plates were tapered down to ⅝ inch width, so that only the top horizontal flow deflector plate spans most of the annulus. This reduced the flow velocities seen at the bottom of the core.

Three lifting lugs are welded on the inside of the CSS top flange. These lugs permit lifting the CSA out of the core when required, such as for vessel inspections.

2.5 Core Barrel Assembly

The core barrel assembly (Figure 2-11) consists of a core barrel cylinder, top and bottom flanges, former and baffle plates, and a thermal shield cylinder. The bottom flange of the CSS is bolted to the top flange of the core barrel cylinder and the lower internals assembly bolts to the core barrel cylinder bottom flange. Its functions are to direct the coolant flow and to support the lower internals assembly. It also reduces the amount of radiation that reaches the reactor vessel (thermal shield).

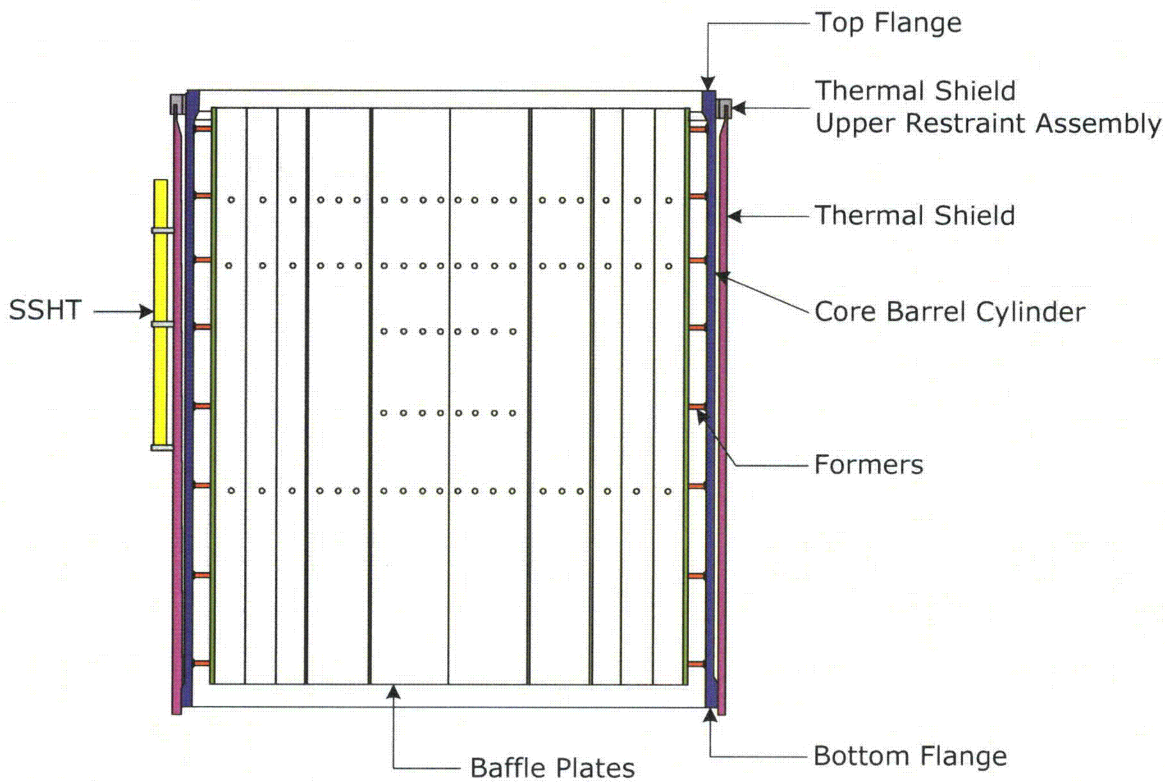

Figure 2-11
Core Barrel Assembly

The original design for all of the plants also included SSHTs attached to the outer surface of the thermal shield. The function of the SSHT is to provide positioning and support for irradiation specimens. In the mid-1970s, degradation of these assemblies led to partial or complete removal of the original design. At CR-3 and DB, the SSHTs were redesigned and are the only remaining B&W-design plants with functional SSHTs.

Incoming cold RCS flow is directed downward along the outside of the core barrel assembly and upward through the fuel assemblies contained inside the core barrel. A small portion of the coolant flows upward through the space between the core barrel cylinder and the baffle plates. A small portion of the coolant also runs down the annulus between the thermal shield and the core barrel cylinder, through holes drilled in the core barrel bottom flange, and then up through the core.

The core barrel is a cylinder approximately 12¼ feet high and two inches thick. It is formed from two rolled plates, and therefore has both vertical and circumferential welds. The core barrel top and bottom flanges are welded to the ends of the cylinder. The CSS assembly bolts to the top flange with 120 bolts secured with locking clips or locking cups. The bottom flange has 30 ¾-inch holes drilled in its side to provide a flow path from the annulus between the thermal shield and the core barrel cylinders to the core. The lower grid assembly is bolted to the core barrel bottom flange with 108 bolts secured with locking clips or locking cups (except for ONS-1, which has 12 additional bolts over the guide lug locations).

The vertical baffle plates form an outer perimeter of the core area to confine and direct the flow of reactor coolant. The baffle plates do not ordinarily provide any structural support to or affect the alignment of the fuel assemblies since there is a clearance between the outer fuel assemblies and the baffle plates. The baffle plates are approximately ¾-inch thick, with widths varying from about eight to 45 inches. There are ¼-inch flow slots between the baffle plates. These flow slots span the third to fifth former plate elevations. The flow slots, along with rows of 1⅜-inch diameter flow holes drilled through the baffle plates at various elevations, minimize the pressure drop across the baffle plates. At seven elevations, the baffle plates are bolted to the formers with 756 bolts secured in place by stainless steel locking pins. At the vertical joints where two baffle plates meet to form corners, a total of 612 bolts secured with locking rings hold the plates together. Figure 2-12 is a sketch showing the inside of the core barrel with the baffle plates and one representative fuel assembly in place.

The 1¼-inch thick former plates provide horizontal framing to support the vertical baffle plates at eight elevations. The outside edges of the formers curve to match the inside surface of the core barrel cylinder to which they are fastened with a total of 704 cap screws held in place with locking pins. At 16 locations on the top and bottom rows of formers, Alloy X-750 dowels are used to locate the formers on the core barrel cylinder. Inside surfaces of the formers are either flat or step shaped to support the various baffle plates. The formers have small holes to permit some reactor coolant to flow up through and cool the spaces between the baffles, formers, and core barrel cylinder.

At the fourth elevation from the bottom, near the hottest section of the core, the ring of former plates are narrower than those at the other elevations, and the baffle plates are bolted to these narrower formers with special screws (secured in place with dowels) that maintain a ¼-inch gap between the baffle plates and former plates. This arrangement provides additional cooling flow to the hottest portion of the baffle plates and some flexibility to the assembly.

There are 20 thermal shield upper restraint assemblies used to bolt the upper end of the thermal shield to the outer wall of the core barrel cylinder top flange. Each assembly consists of three rectangular blocks that are bolted together. The inner block, the shim, serves to keep the assembly at the correct distance out from the core cylinder wall. The inner "B" and outer "A" blocks are recessed at the bottom, such that a slot is formed (after assembly) to provide radial restraint at the top of the thermal shield, while allowing axial thermal growth relative to the core barrel and CSS. Each assembly is fastened together with two cap screws bolted from the shim side. The restraint assemblies are then positioned and secured to the core barrel and thermal shield with three dowels (captured by welded plugs), and three restraint bolts secured with locking clips welded to the restraints. The lower end of the thermal shield is shrunk fit on the lower grid flange and fastened by 96 bolts or studs and nuts secured with locking clips or locking cups.

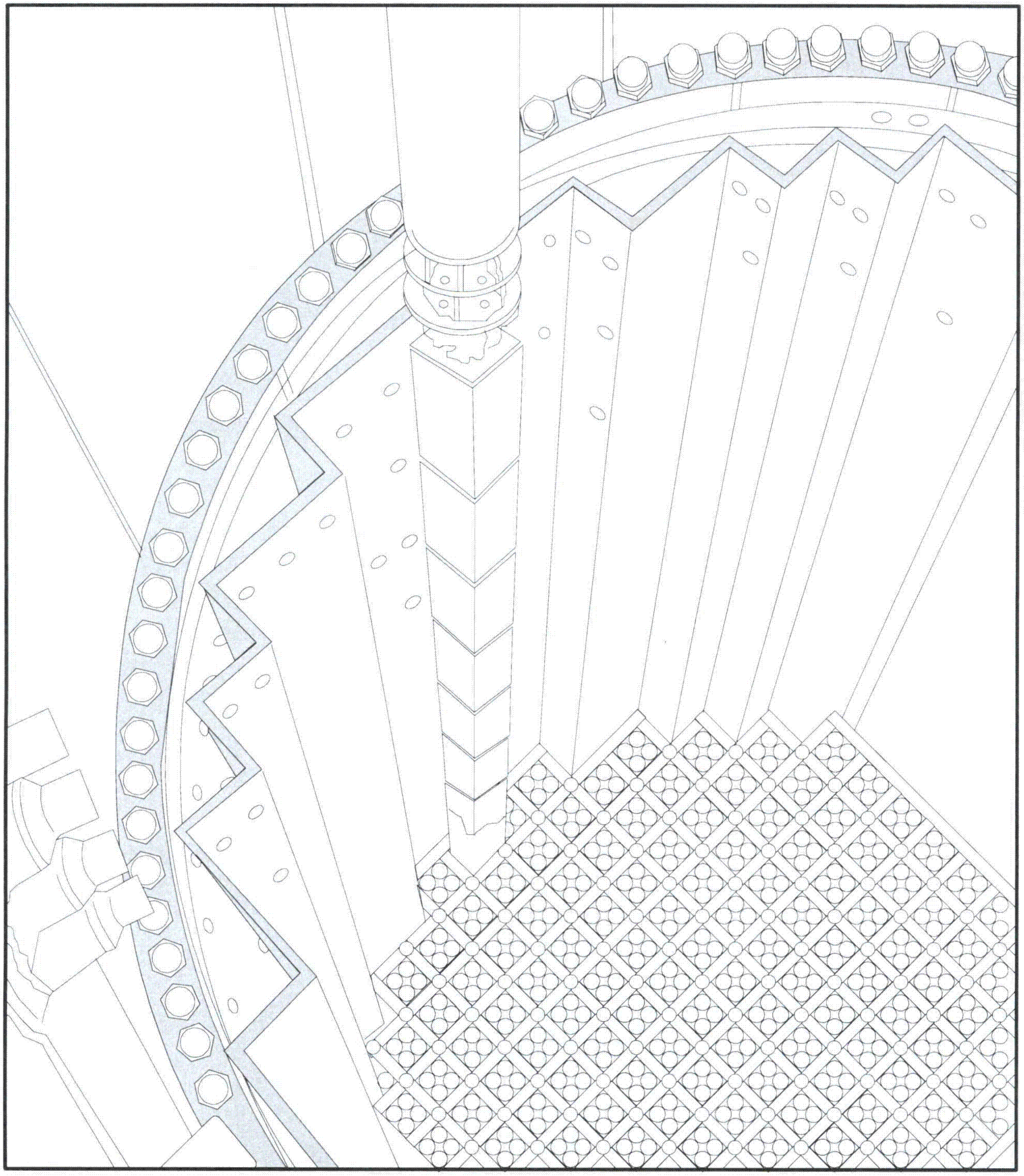

Figure 2-12
Core Barrel Interior Schematic

2.6 Lower Internals Assembly

As shown in Figure 2-13, the lower internals assembly consists of a lower grid assembly, a flow distributor assembly, and incore monitoring instrumentation (IMI) guide tube assemblies. The lower grid assembly is a series of grid and support structures bolted to the bottom of the core barrel to provide structural support to the core. The flow distributor assembly is a set of flow distribution plates, located below the lower grid assembly, which helps direct coolant flow upwards towards the core. The IMI guide tube assemblies run through and are supported by both the flow distributor and the lower grid assemblies, and provide support and protection for the IMI.

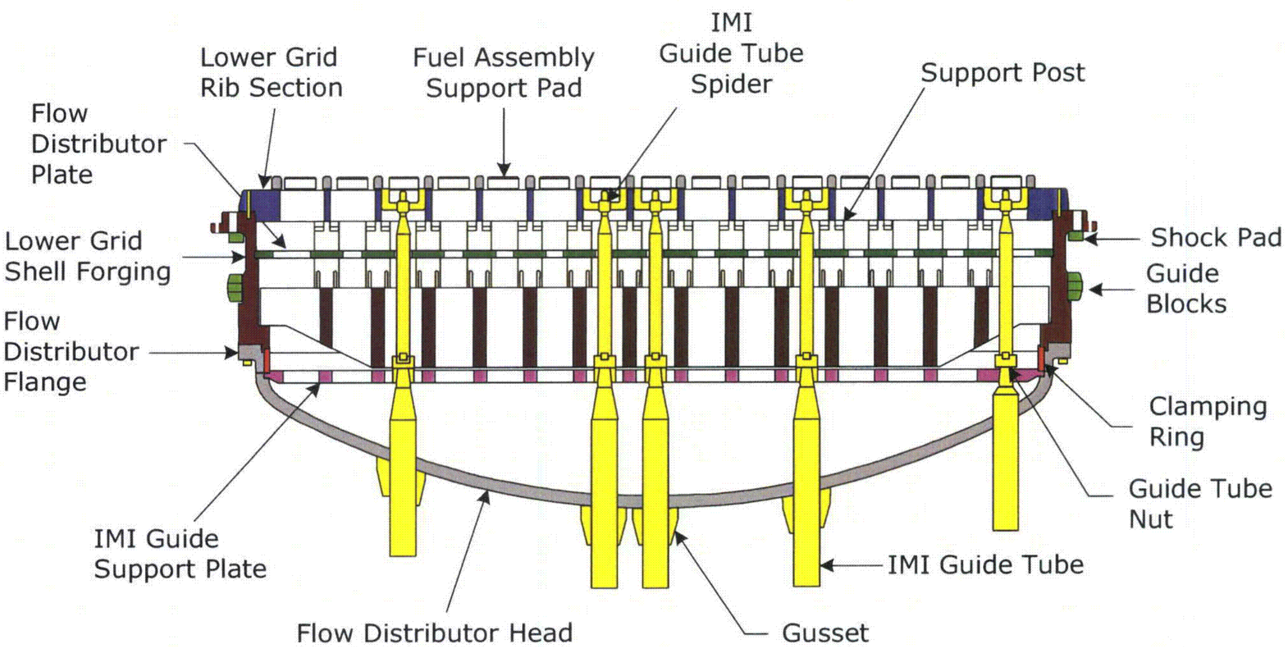

Figure 2-13
Lower Internals Assembly

2.6.1 Lower Grid Assembly

The lower grid assembly provides alignment and support for the fuel assemblies, supports the core barrel assembly and flow distributor, and aligns the IMI guide tubes with the fuel assembly instrument tubes. The lower grid consists of three grid structures or flow plates. From top to bottom they are the lower grid rib section, the flow distributor plate, and the lower grid forging. Each of these flow plates has holes or flow-ports to direct reactor coolant flow upward towards the fuel assemblies. The lower grid assembly is surrounded by the lower grid shell forging. The lower grid shell forging is a flanged cylinder ("ring"), which supports the various horizontal grid structures and flow plates.

The lower grid rib section is a five-inch thick, 141-inch diameter disk through which 177 squares are machined out, leaving a grid with 1-inch wide "ribs." The square holes align with the fuel assembly locations in the core. There are additional holes about the periphery of the disk to permit a small bypass flow of reactor coolant up behind the baffle plates in the core barrel.

There are 384 small fuel assembly support pads attached to the top of the rib section to provide a seating surface and support for the bottoms of the fuel assemblies. A cap screw is used to hold each pad in place. Two Alloy X-750 dowels position each pad. Below the rib plate at 48 grid intersections, there are support post assemblies that provide support from the lower grid forging. The support post assemblies are fastened in place with cap screws secured with welded locking pins.

The spider castings are cylinders with four legs that are welded to the walls of 52 of the holes in the lower grid rib section to provide support for the tops of the IMI guide tubes.

The lower grid flow distributor plate, located midway between the lower grid rib section and the lower grid forging, aids in distributing coolant flow. It is a flat one-inch thick, 135⅞-inch diameter perforated plate with a ⅛-inch lip around the bottom. The flow distributor plate rests on and is welded to a ½-inch lip on the lower grid shell forging.

The flow distributor plate has 677 3⅜-inch diameter flow holes (177 of which are aligned with the center of the fuel assemblies). Twelve of the normal flow holes near the center of the flow distributor plate are fitted with orifice plugs, which reduce the diameter of the flow port down to 1⅞ inches. There are also 24 smaller flow holes and 48 holes to accommodate the support posts. The support posts are welded to the flow distributor plate.

At all plants except ONS-1, the lower grid forging is a single 135-inch diameter forged disk that serves as the main weight-bearing structure in the lower grid. The majority of the lower grid forging, i.e., the center 96 inches of the disc, is 13½-inches thick. The disc tapers to six inches thick at its edges. There are 177 flow holes machined out of the lower grid forging, aligned with the fuel assemblies above. The lower grid forging is welded to the lower grid shell forging. At ONS-1 only, the lower grid forging is fabricated as a lattice grid from ribs, similar to the plenum cover weldment described in Section 2.3.1. The lower ends of the 48 support post assemblies are welded to the top of the lower grid forging.

The lower grid shell forging is a two-foot high, 136-inch ID cylinder with numerous internal and external flanges and lips that support the various items of the lower grid assembly. The lower grid shell forging is four inches thick at its thinnest cross-section.

The lower grid shell forging is bolted to the core barrel lower flange with 108 core barrel bolts, described previously. The lower end of the thermal shield is shrunk fit on the lower grid flange and fastened by 96 bolts, or studs and nuts, secured with locking clips or locking cups. The lower grid rib section is fastened to the shell forging with 36 cap screws secured with welded locking pins. The flow distributor plate rests on and is welded to a ½-inch lip on the lower grid shell forging. The lower grid forging rests on and is welded to the top surface of the lower grid shell forging lower flange. The flow distributor assembly bolts to the bottom of the lower grid shell forging with 96 bolts secured with locking clips. The lower surface of the bottom flange of the lower grid shell forging holds the clamping ring in place, which holds the IMI guide support plate in place against the flow distributor flange.

Guide blocks are bolted at 12 equidistant azimuthal locations around the outside vertical wall of the lower grid shell forging. These blocks are machined after trial fit-up of the CSA into the RV in order to provide precision clearances with the sides of the guide lugs welded to the wall of the RV.

The 24 guide blocks are each 6½ inches wide, five inches high with beveled guiding/mating surfaces extending out three inches from the shell forging wall. Each is held in place with a bolt and washer and an Alloy X-750 dowel.

Twelve shock pads are bolted to the lower surface of the upper flange of the lower grid shell forging, located directly above the reactor vessel guide lugs. In the event of a core barrel joint failure, the RV core guide lugs and lower grid shock pads will limit the core drop to approximately ½ inch.

The support posts are 48 cylinders placed between the lower grid forging and the lower grid rib section to provide support. The support post assemblies consist of the support pipes and the associated bolting plugs. The support pipes are made from 10½-inch high sections of four-inch schedule 160 pipe. There are four equally spaced notches at the bottom of the cylinders, where they are welded to the top of the lower grid forging that allow coolant flow upward from below. The bolting plugs are 1¾-inch high disks welded to the top of the support pipes. The bolting plugs have four scallops shaped holes machined out of the edges so that the tops have a cruciform shape through which coolant can flow. The top of each bolting plug is drilled and tapped to accept the cap screw used to hold it to the lower grid rib section.

2.6.2 Flow Distributor Assembly

The flow distributor assembly supports the IMI guide tubes and directs the inlet coolant entering the bottom of the core. It consists of a perforated head (plate), a flange, an IMI guide support plate, and a clamping ring.

The flow distributor head is a two-inch thick, 136-inch ID bowl-shaped plate that bows downward about 20 inches. The head is welded to the flow distributor flange, which is five inches high, with an approximately three-inch thick flange extending out to a 142-inch OD. The IMI guide support plate fits across the flange, resting in a lip in the flange. The four-inch high, one-inch thick clamping ring fits against the inside diameter of the flange on top of and holding the IMI guide support plate in place. This whole assembly is bolted to the bottom of the lower grid shell forging with 96 bolts secured with locking clips.

There are 52 approximately 4½-inch diameter holes through which the IMI guide tubes pass. Fifteen of these holes have shallow counterbores on the bottom edge to permit welding the IMI guide tubes directly to the flow distributor head plate. The remaining 37 guide tubes are secured by a set of four gussets, which are ¾ inch thick triangular shaped pieces, six inches high and 1¾ inches wide. The long sides of the gussets are welded to the IMI guide tubes and the bases are welded to the flow distributor head. There are 156 six-inch diameter holes and five 3½ inch diameter holes in the flow distributor head to permit reactor coolant flow upward through the lower grid assembly.

The IMI guide support plate is a 134-inch diameter, two-inch thick disk, with 52 shaped holes to accommodate the IMI guide tubes. The IMI guide tubes are held in place by washers and guide tube nuts secured by welded locking clips. At 46 of the holes, there are also four oval-shaped flow ports machined through the IMI guide support plate to permit reactor coolant flow parallel to the tubes. There are also numerous holes between 6½ and 7½ inches in diameter for reactor coolant flow.

2.6.3 IMI Guide Tube Assemblies

The IMI guide tube assemblies guide the 52 IMI assemblies from the IMI nozzles in the RV bottom head to the instrument tubes in the fuel assemblies. Horizontal clearances are provided between the IMI nozzles and the IMI guide tubes in the flow distributor to accommodate misalignment. The IMI guide tubes are designed so they will not be affected by core drop.

The IMI guide tubes are long tapered tubes through which the incore nuclear detectors and thermocouples are fed up into the fuel assemblies. The diameters vary along the length of the IMI guide tubes. At the top, where they are held in place by the spiders welded into the lower grid rib section, the IMI guide tubes have a one-inch OD with a 0.60- to 0.67-inch center bore. At the bottom, the IMI guide tubes have a 4½-inch OD with a 3½-inch ID. The top 32 inches of all 52 IMI guide tubes, from where they penetrate the flow distributor up to the spiders in the lower grid rib section, are essentially identical. There are ten different IMI guide tube models, however, which differ in their overall length, varying from 77¾ to 51¼ inches. The length required depends upon the location within the core, as the distances vary between the IMI guide support plate and the flow distributor head and between the flow distributor head and the bottom of the RV.

The IMI guide tube assemblies are attached to the bottom of the flow distributor head either by a weld bead around the full circumference of the IMI guide tube, or by four gussets that are welded to the flow head and the IMI guide tubes. The IMI guide tubes then have an interference fit through holes in the IMI guide support plate. The IMI guide tubes are held to the top of the IMI support plate with washers and the guide tube nuts. The outside of the IMI guide tubes have a 1¾-inch section of threading at this location to engage with the guide tube nuts. The IMI guide tubes have an approximate two-inch diameter where they pass up through 6½-inch diameter holes in the lower grid forging and the 3⅜ inch diameter holes in the flow distributor plate.

The guide nuts are 2½-inch tall, ½-inch thick nuts that fit over the IMI guide tubes and secure them to the top of the IMI support plate. The guide nuts are secured with locking clips.

Spider castings are welded in 52 of the holes to provide support for the IMI guide tubes. The spider castings are 1¾-inch high, one-inch ID cylinders with four ¼-inch thick L-shaped legs, which extend out to and are welded to the walls of the holes in the lower grid rib section. The inner diameters of the spider tube cylinders are chrome plated 0.0002 to 0.0004 inches thick. The chrome-plated bore of the spider hub forms a guide bushing for the top of the IMI guide tube assembly to accommodate longitudinal thermal expansion.

3 FAILURE MODES, EFFECTS, AND CRITICALITY ANALYSIS (FMECA)

The objective of this analysis is to provide a systematic, qualitative review of the B&W-designed PWR internals to identify combinations of internals component items and age-related degradation mechanisms that potentially result in degradation leading to significant risk. The FMECA is used to examine the susceptibility, and safety and economic consequences of identified internals component item/age-related degradation mechanism combinations. The scope of the FMECA is limited to PWR internals, although the FMECA (and the results) refer to associated component items, e.g., reactor vessel guide lugs. There are no specific FMECA entries for these associated component items.

3.1 Analysis Approach/Source Information

The FMECA approach uses inductive reasoning to ensure that the potential failure of each component item is analyzed to determine the results or effects thereof on the system and to classify each potential failure mode according to its severity. The FMECA approach is very flexible and can be adapted in many different ways to accomplish a variety of purposes. For example, a FMECA can be used to contribute to improved designs, to establish and prioritize maintenance plans for repairable systems, as a resource in troubleshooting efforts, or as a training tool for new engineers.

The first step in performing a FMECA is to define the system under investigation. Then, an appropriate level of detail is selected within the system (e.g., components, subsystems). For this FMECA, the level of detail is internals component items. Next, all "component items" at the identified level of detail are enumerated to produce a mutually exclusive and complete rendering of the entire "system" under study. For each component item, a complete set of failure modes is specified, and the effect(s) of each failure mode on the system is determined. This information is placed on a FMECA table (see Appendix A). The headers for the FMECA table columns are discussed and defined in Section 3.2. The next step is for each failure mode to be judged on its importance to risk, based on the susceptibility (likelihood of the degradation mechanism) and severity of consequences. For this FMECA, consequences were examined from two perspectives: safety and economic; the criticality metrics used were qualitative and also defined in Section 3.2.

The B&W-designed PWR internals component item names were generally taken from previous generic license renewal work [7] and MRP-sponsored IMTs [6, 8]. The age-related degradation mechanisms considered in the FMECA were taken directly from the results of the component item screening task as documented in Reference [9]. The local and global effects were based on engineering judgment, supported by engineering drawings (showing the relationship between internals component items, weld locations, etc.), internals photographs, etc. The development of the criticality metrics and the ability to detect (or not detect) the failure mode were derived from the expert panel meeting (see Section 3.3). A risk matrix was developed to permit assignment of the FMECA results into risk bands (see Section 3.5). Figure 3-1 provides a flowchart of the FMECA process that is discussed in the subsequent subsections.

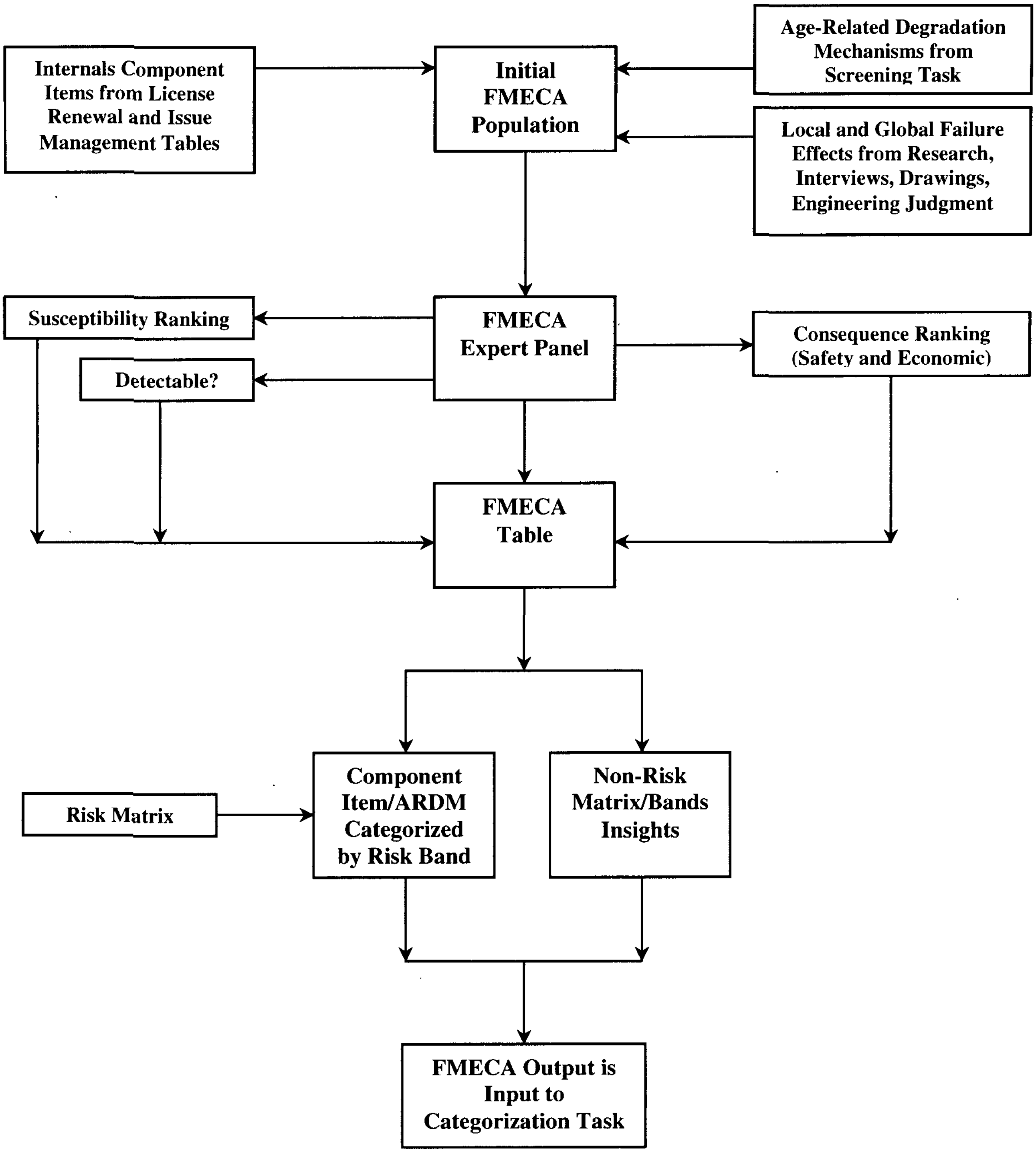

Figure 3-1
FMECA Process Flowchart

3.2 FMECA Table

The FMECA is performed by completing the columns of the FMECA table for each component item of the PWR internals considered. A FMECA table is developed for each of the four internals "assemblies," namely:

- Plenum Assembly
- Core Support Shield Assembly
- Core Barrel Assembly
- Lower Internals Assembly

These tables are contained in Appendix A of this document. The plenum assembly is further subdivided in Table A-1 as: plenum cover assembly, plenum cylinder assembly, upper grid assembly, and control rod guide tube assembly. Similarly, the lower internals assembly is further subdivided in Table A-4 as: lower grid assembly, flow distributor assembly, and IMI guide tube assemblies. The definition for each FMECA table header and the guidelines for populating each column are discussed below.

Component Item Name

This column contains the name or nomenclature of the PWR internals component item being analyzed. High-level functions for each assembly (or sub-assembly) are provided in the FMECA table prior to the listing of all the individual component items. Some component items are listed with a "simple" name, e.g., dowel, which may appear more than once in a particular table or in several tables. However, such component item names are italicized to indicate that those component items are associated with the immediately preceding component item in the table. For example, the weldment rib pads (Table A-1) are associated with the weldment ribs that immediately precede its entry in the FMECA table. Typically, dowels and locking devices are italicized in the tables.

FMECA Identifier

The FMECA identifier is an internal FMECA "label." These can be used to facilitate cross-referencing to other parts of the FMECA and to facilitate finding component items. The identifiers are structured on a per-assembly basis as follows:

- Plenum Cover Assembly P.1.x
- Plenum Cylinder Assembly P.2.x
- Upper Grid Assembly P.3.x
- Control Rod Guide Tube Assembly P.4.x
- Core Support Shield Assembly S.x
- Core Barrel Assembly B.x

- Lower Grid Assembly L.1.x
- Flow Distributor Assembly L.2.x
- IMI Guide Tube Assemblies L.3.x

Some FMECA identifiers have a "letter" appended, e.g., B.4a, B.4b, B.4c, B.4d. The "letters" are the same component item, but repeated in the table. The differences in these "same" component items are generally material differences at different plant sites. Material information (e.g., Alloy A-286 versus Alloy X-750) is not generally given in the FMECA tables, but is provided in Reference [8].

Degradation Mechanism

The age-related degradation mechanism that can potentially affect the listed internals component item is identified in this column. The age-related degradation mechanisms considered in the FMECA were taken directly from the results of the component item screening task as documented in Reference [9]. A particular component item may be subject to one or more mechanisms. For some component items, there are no identified age-related degradation mechanisms; these are identified with "no credible degradation mechanism." The screening criteria discussed in MRP-175 [3] to distinguish between "Category A" and "Non-Category A" were used to determine which age-related degradation mechanisms would be considered in the FMECA; the actual screening process is reported in Reference [9]. The degradation mechanisms that could be screened in for PWR internals were listed in Section 1.2.

Failure Mode

This column provides how the age-related degradation mechanism will affect the identified component item.

Failure Effects (Local Effects)

The consequence of each assumed failure mode on component item's operation, function, or status is identified, evaluated, and recorded within the FMECA tables. The local failure effects concentrate specifically on the impact that an assumed failure mode has on the operation and function of the component item under consideration.

Failure Effects (Global Effects)

Each local effect has the potential to impact the "system" with a global impact. Often there is no operational effect as a result of the local effect, such that the plant can and will continue to operate as normal. See Section 3.4 below for a discussion of common cause failure (CCF) and cascading (or dependent) failures.

Criticality Metrics

The criticality metrics of a particular component item failure are evaluated qualitatively by assessing both the susceptibility to an age-related degradation mechanism and the severity of the consequences. For this FMECA, two types of consequences are considered: safety and economic. When considered together, the criticality metrics represent the risk due to the failure of a particular component item (see Section 3.5).

Susceptibility

The susceptibility metric is a qualitative assessment of the likelihood (expressed as a probability or frequency) that an age-related degradation mechanism might occur, given the existing environmental conditions (e.g., temperature, pressure, and fluence), material properties (type of metal, stress-strain), etc. occurring over the life of a nuclear power plant (up to 60 calendar years, considering license renewal). The susceptibility is unrelated to the consequences, e.g., the component item failure or loss of function. The susceptibility qualitative metric was determined as a result of the expert panel meeting. This criticality metric uses an A, B, C, D scale (increasing frequency).

A – Improbable: not likely to occur ("Category A" from the screening task is synonymous with this susceptibility metric)

B – Unexpected: not very likely to occur, though possible; conditions are such that the age-related degradation mechanism is not expected to occur very often

C – Infrequent: likely to occur, conditions are such that the age-related degradation mechanism is expected to occur occasionally

D – Anticipated: very likely to occur; conditions are such that the age-related degradation mechanism is expected to occur

The susceptibility is sometimes modified with an "I" to indicate an improbable occurrence over the 60-year time period being considered. For example: B/I indicates an unexpected, but possible, degradation mechanism whose initiation results in a certain state that is not credible (or improbable), e.g., SCC crack leading to a 360 degree weld crack. To carefully distinguish between the different types of likelihood, it is possible (B) to have SCC cracking around a weld, but improbable (I) that such as crack would grow around the weld to the critical crack size needed to fail the weld. Component item/degradation mechanism pairs identified as improbable are not explicitly evaluated for consequences. However, consequences can often be inferred from the local and global effects of related or similar component items. Those items identified as improbable, but which will either result in severe consequences, affect the ability to cope with a LOCA, or will require the successful "operation" of the guide lugs, are bolded in the FMECA table, and will be called out separately in the results as items that should be considered for inclusion in an inspection program.

Severity of Consequences

Severity classifications are assigned to provide a qualitative measure of the potential consequence resulting from a component item failure. For those component item/age-related degradation mechanism pairs for which the susceptibility metric was assigned an "A," i.e., "Category A," there was no subsequent evaluation of the consequence due to the very low (i.e., improbable) event frequency. For the PWR internals FMECA, two aspects of consequences are considered: safety and economic. Thus, there are two columns in the FMECA for which qualitative metrics are assigned. The two sets of severity of consequence qualitative metrics were determined as a result of the expert panel meeting. These criticality metrics use a 1, 2, 3, 4 scale (increasing severity).

For severity of consequences (safety), the qualitative metric has been defined as:

1 – Safe: no or minor hazard condition exists

2 – Marginal: safe shutdown is possible (though with reduced margins to adequately cool the core and/or successfully insert the control rods); localized fuel assembly damage

3 – Severe: safe shutdown is possible (though with very reduced margins to adequately cool the core and/or successfully insert the control rods); core damage (multiple damaged fuel assemblies)

4 – Critical: safe shutdown is not possible (margins to adequately cool the core and/or successfully insert control rods are totally eroded); extensive core damage

The safety consequence metric assigned will be the highest value, i.e., bounding consequence, for normal operation or design basis event (transient, LOCA, seismic) when the failure mode is not detectable. Typically, the safety consequences were estimated to be the same for normal operation and a design basis event (when the failure mode is not detectable).

For severity of consequences (economic), the qualitative metric has been defined as:

1 – No or trivial cost

2 – Cost that can be generally handled within the existing plant budget and resources

3 – Cost that exceeds the normal plant budget and resources

4 – Cost that potentially affects the utility's overall financial health

Note that the economic consequences assume that the failure mode is discovered through some means, e.g., plant inspection or notification of discovery at another plant site. This is also conservative when assessing the risk.

Detectable

This column provides information about whether or not the failure mode (and subsequent local/global effects) is detectable. "Detectable" has been defined as answering "yes" to either of the following questions:

Can and would the operators, through normal instrumentation, detection systems (e.g., neutron noise or loose parts monitoring), and surveillance and response procedures, be aware of the failure mode either directly or indirectly, e.g., via a consequential event such as a loose part, and take an appropriate action?

Can the operators be aware of the failure mode during the normal activities that occur during a refueling outage (e.g., lifting of the plenum assembly), and take an appropriate action?

When the answer is "no," the FMECA safety consequence must consider the impact of a design basis event (e.g., seismic, LOCA, transient). Detectability does not rely on "external" periodic inspection programs, but rather on the normal, observable plant responses related to the effect of the aging degradation mechanism.

Comments

This column contains, as needed, any addition information or pertinent remarks pertaining to and/or clarifying any other column in the FMECA. Comments may also include a notation of unusual conditions, failure effects of redundant component items, recognition of particularly critical design features, or any other remarks that amplify the line entry.

3.3 Expert Panel

To facilitate populating the FMECA table, an expert panel was convened. The participants, covering a wide spectrum of technical acumen, are identified with their area of expertise in the acknowledgements. Because some component items have multiple functions, and different failure modes that may lead to a variety of consequences, this group met for a day-and-a-half to discuss each internals component item and the associated degradation mechanism(s).

The meeting began with an explanation of the purpose of the FMECA in the context of the entire MRP effort to develop a categorization process, and ultimately an inspection strategy. A brief explanation of the FMECA method was provided, as well as defining each of the columns. In particular, the qualitative metrics for susceptibility, severity of consequences (safety), and severity of consequences (economic) were defined and discussed.

As the meeting started, paper copies of the FMECA table were provided with all but the criticality metrics columns completed. There were several stated objectives of the FMECA expert panel meeting; these were:

- verify the appropriate local and global effects for each degradation mechanism
- verify the "Category A" component items, listed in the FMECA as "no credible degradation mechanism"
- fill in the criticality metrics columns
- determine if the failure mode was detectable

During the meeting, it was recognized that if the failure mode was not detectable, a second consequence question needed to be posed: would the degradation mechanism result in a more severe consequence (if undetected) when a design basis event occurred (e.g., seismic or LOCA). The consequence column metric is the most conservative consequence (between normal operation and consideration of a design basis event, when needed).

After the meeting, it was recognized that in the "same as another component item" process that took place during the meeting that some of the economic consequences were probably under-estimated, particularly when there was a possibility of fuel damage, and in some cases not evaluated consistently. Changes that were made to the FMECA were highlighted and distributed to the expert panel participants to ensure that the intent of the group was not distorted. Other editorial changes were made to enhance clarity and readability.

3.4 Other Issues

There are some other issues related to failure of PWR internals component items that are not easily reconciled with the analysis approach of a FMECA. Since a FMECA structure essentially forces the analyst to evaluate failure modes on a component item-by-component item basis, a FMECA is not a tool that typically or effectively addresses the issues of common cause failures and cascading (or dependent) failures.

3.4.1 Common Cause Failures (CCF)

Since a FMECA tends to focus on a single component item and a particular failure mode (at a time), it was not expected that the PWR internals FMECA would be able to systematically identify and evaluate CCFs. However, CCF is not likely to be of a significant concern for internals affected by age-related degradation mechanisms.

IASCC, SCC, wear, fatigue, thermal aging embrittlement, irradiation-induced embrittlement, thermal and irradiation-enhanced stress relaxation and creep, and void swelling are potential material degradation mechanisms for PWR internals component items. All these mechanisms require a material that is sensitive to the aging degradation and an environment conducive to the degradation for PWR internal component items to exhibit degradation. For example, for SCC and IASCC, reactor coolant combined with tensile stress are required for these mechanisms to operate. IASCC also requires high neutron fluence. Void swelling requires high neutron fluence and temperature above approximately 320°C (608°F). Thermal aging embrittlement of cast austenitic stainless steel and other susceptible alloys requires exposure to operating temperature and above for significant periods of time. Irradiation-induced embrittlement requires high neutron fluence.

The relative susceptibility of internals component items within a material type (wrought austenitic stainless steel, cast austenitic stainless steel, age-hardenable stainless steel, nickel-based alloys, etc.) to any of these aging degradation mechanisms depends on the variability of chemical composition, forming process, welding process, heat treatment, and the resultant microstructure of the material. These parameters are controlled to within an acceptable band, but are otherwise nearly random. Therefore, degradation of internals component items caused by any of these mechanisms is a stochastic process, that is, a random process, with the expectation that initiation will occur at different times and growth will proceed at different rates; hence these age-related mechanisms will not lead to common cause failures of PWR internals component items. For example, all Alloy A-286 core barrel bolts with the same forming process, chemical composition, and heat treatment will not exhibit cracking and failure within a small time interval causing the downward displacement of the core. Inspection of the core barrel bolts will identify cracking in the highly susceptible bolts before subsequent failure of the remaining bolts.

Nonetheless, with sufficient time, though improbable, all "like" component items could eventually be affected by an age-related degradation mechanism. The current inspection process (i.e., visual inspection during a 10-year inservice inspection) would ensure that over the plant's lifetime an excessive number of similar component items (e.g., round bars, shock pads) do not fail from the same mechanism. The FMECA notes some of these improbable occurrences (with an "I" in the susceptibility column). When the consequence of these failures is severe, that metric is bolded in the FMECA table; those component items/degradation mechanisms are separately identified. For example, CCF is not expected to fail all of the round bars on the plenum cylinder and the core barrel cylinder; if, however, all of the round bars do fail, the internals' ability to cope with a LOCA will be severely compromised. Accordingly, these failures are noted in the results (see Section 5), and appropriate monitoring would prevent any multiple failures over time.

3.4.2 Cascading Failures

Cascading or dependent failures is another area that is typically beyond the scope of a FMECA. However, there are some obvious instances that are noted in the FMECA and reported in Section 5. Some of the consequence metric values are marked with an asterisk. As noted in the comments column, the asterisk (*) indicates that the consequences are a result of cascading (or dependent) failures. For example, failure of a core barrel bolted joint, without inspection for cracking, will initiate with a few failures of the most susceptible bolts causing higher loads on the adjacent bolts increasing the progression of SCC and so forth until there is mechanical overload of the remaining bolts. Inspection (as a result of a detection method, e.g., loose parts monitor, neutron noise) of the core barrel bolts will identify cracking in the highly susceptible bolt before the dependent failure of the joint.

3.5 Risk Matrix

A risk matrix was developed to identify risk-significant PWR internals component item/age-related degradation mechanism pairs. Risk for this analysis uses the most basic form of the definition of risk, i.e., the likelihood (of an event) times the consequence (of the event). For the FMECA, the elements of risk have been identified as the FMECA criticality metrics, i.e., likelihood is defined as the susceptibility, and consequence is characterized as the severity of consequences. This "risk metric" is not to be confused with risk in a probabilistic risk assessment, for which the metrics of core damage frequency and large early release frequency are typically used.

The risk matrix is a correlation of the consequence severity of a particular failure mode with the susceptibility of that particular degradation mechanism occurring. The risk matrix was configured such that increased consideration was given to the severity of a particular consequence rather than on its susceptibility. The risk matrix is shown as Figure 3-2. The risk matrix does not include a column for the susceptibility metric value of "A" because, as noted in Section 3.2, the "A" (or Category A) events are deemed so improbable (very, very low likelihood of occurrence) that the severity of consequence metric was not evaluated, implying that even if there was an adverse consequence, the risk impact would be insignificant.

The risk matrix is used to identify component item/degradation mechanisms that expose the internals and the plant to risk. As the risk can vary, different risk bands are considered within the matrix to categorize the level of risk of a particular component item/degradation mechanism pair, and provide guidance on the strategies that should be developed to reduce the corresponding risk. The numbers (developed with engineering judgment) in the risk matrix are used to qualitatively assess the risk in each cell. While these numbers represent no absolute measure of risk, the different weights are meant to represent the increasing change of risk along both dimensions (susceptibility and consequence). The values provide a holistic view of the risk gradients, and are used to develop the five risk bands.

		Increasing Susceptibility →		
		B	C	D
Increasing Consequences ↓	1	1	2	3
	2	5	7	9
	3	9	12	15
	4	13	17	21

Figure 3-2
Susceptibility/Severity of Consequences Risk Matrix

The risk bands within the risk matrix consider five different categories of risk. Each component item/age-related degradation mechanism analyzed will fall within one of the five risk bands. The risk bands are defined as follows:

Risk Band I (risk scores of 0-2)

The risk is not significant. The failure modes with this susceptibility and severity of consequences will have no or minimal risk on the internals or the operation of the plant.

Risk Band II (risk scores of 3-5)

The risk in this band is mild. Failure modes that fall in this band will have a minimal risk impact on the internals or the operation of the plant. Accordingly, some consideration should be given to ensure that these failure modes can be detected at some time in the life of the plant.

Risk Band III (risk scores of 6-9)

The risk in this band is moderate. For failure modes that are in this band, consideration should be given to a strategy to ensure that detection (and possibly mitigation) exists.

Risk Band IV (risk scores of 10-15)

This risk in this band is significant, and accordingly, degradation mechanisms need to be discovered before the consequences are ever realized.

Risk Band V (risk scores of greater than 15)

The risk is so high in this band that an immediate re-evaluation of the design is necessary. Since redesign of the PWR internals is not a viable option, early detection of such degradation mechanisms is important to prevent the consequences from ever being realized.

4 KEY ASSUMPTIONS

Several assumptions and observations have been made during the development of the FMECA; these include:

- The FMECA was performed for a bounding, generic B&W-designed PWR internals. While there are differences in internal designs (between plants) noted in the FMECA table, when consolidating the results into risk bands, only the most conservative risk was used for like component items, e.g., original and replacement barrel bolts.
- In general, the FMECA table was developed assuming a normal, at-power, steady-state operation of the plant. Design basis events were only considered if there was no means to detect a failure mode. In these cases, only the impact on safety consequences was considered.
- The FMECA only considered failure modes that resulted from age-related degradation mechanisms (as listed in Section 1.2) known to potentially occur in PWR internals. There was no consideration given to manufacturing errors, maintenance errors, installation errors, transport errors, or any other type of random or human errors.
- As discussed in Section 3.4, CCFs and cascading failures were not systematically evaluated in the FMECA. Nonetheless, obvious and risk-significant occurrences of both were identified and included in the results (Section 5).
- Wear and fatigue are generally coupled with irradiation-enhanced stress relaxation and creep since it is assumed that wear and fatigue are a direct consequence of such stress relaxation and creep, as indicated in the screening table in Reference [9]. There are instances of wear not induced by irradiation-enhanced stress relaxation and creep.
- Consequence of Failures (COFs) as defined in the IMT [6,8] were not used as input to the FMECA. The COFs were a high-level mechanism for defining the type of consequence that would be experienced from a component item's failure; the local and global effects considered in the FMECA were more detailed. The limitation of the IMT work was noted in the MRP-156[3]. In Appendix B, a qualitative comparison between the IMT approach and the FMECA is provided for reference.

These assumptions are either bounding or methodological, and do not require plant-specific verification for each of the B&W-designed operating units.

[3] "It is important to note that the information presented in this report (MRP-156) is focused on Phase Two of a three-phase effort. It is not intended to support reliability assessments of plant-specific evaluations. Prior to implementation of Phase Three, the consequence of failure needs to be further evaluated by considering the severity and frequency of the failure. When combined, these define the risk associated with loss of function of the evaluated components."

5
SUMMARY OF RESULTS

This section discusses the results of the FMECA as they pertain to the safety and economic risks associated with the B&W-designed PWR internals. The overall results are presented in the FMECA tables in Appendix A, where an individual entry for each combination of internal component item and age-related failure degradation mechanism is provided. As described in Appendix A, there are four FMECA tables, one for each of the major internals assemblies, i.e., plenum assembly (A-1), core support shield assembly (A-2), core barrel assembly (A-3), and lower internals assembly (A-4). These tables are populated according to the discussion of each column header in Section 3.2.

The information in the FMECA tables are "processed" by using the risk matrix developed in Section 3.5. Two risk matrices are used: one to portray the safety risk, and the second for the economic risk. Table 5-1 summarizes the results of the FMECA for both the safety and economic consequences. The susceptibility metric listed in the FMECA is used for both consequence types. For each internals component item/age-related degradation mechanism considered in the FMECA, two risk "scores" were determined from the criticality metrics columns, one for each consequence type. For example, for item P.1.1, the two risk "scores" were B1 (safety) and B2 (economic). The risk "scores" were then assigned a risk band, as defined in Section 3.5. In this case, the risk "scores" would be in Risk Band I and Risk Band II, respectively. The final tallies (for all four internals assemblies) are provided in Table 5-1.

Table 5-1
Summary of Risk Matrix Results for Safety and Economic Consequences

Risk Band	Safety(a)	Economic(a)
I	124	30
II	21	77
III	18 + (6)	53
IV	(3)	3 + (6)
V	0	(3)

(a) The number in parenthesis (#) indicates risk as a result of a cascading failure and not as a direct consequence of a particular age-related degradation mechanism.

In determining the tallies reported in Table 5.1, any component items with FMECA identifiers with appended letters, e.g., B.4a, B.4b, were counted as a single, bounding, risk "score." Failure modes that were identified as improbable are not explicitly counted in any risk band. (These failures, however, are discussed below.) Also, for component items/degradation mechanisms that were assigned the lowest susceptibility (i.e., improbable), no consequences were evaluated. Accordingly, there are 34 "A" pairs that are not included in the risk matrix summary. Finally, the identified cascading failures are included in the tally, though marked and distinguished from the other failure modes because it is recognized that time (and no inspection program) is an important factor for the cascading failures to occur. Nonetheless, they are included as some level of inspection (to be determined) is necessary to ensure that these cascading failures never occur.

Without the cascading failures, Table 5-1 shows there are 17 internals component items/age-related degradation mechanism pairs that are in the moderate safety risk band (Risk Band III). This is not surprising considering the level of redundancy inherent in the internals design. In general, the age-related failure of an internals component item is not a safety risk concern. Equally expected, the cascading failures are in the moderate and significant safety risk bands (Risk Bands III and IV). The specific pairs are listed in Table 5-2, and show the three sets of connecting bolts. Table 5-3 summarizes the safety consequence results for Risk Band III, showing flange wear failures, baffle and former plate and connector failures, as well as six cascading failures.

Tables 5-2 through 5-6 were populated by identifying the susceptibility (metric) and the severity of consequence (metric) from the Tables in Appendix A. This pair of metrics was used with Table 3-2 to determine what risk band was associated with the component item/age-related degradation mechanism. This activity was performed for both safety and economic severity of consequence metrics.

Table 5-2
Summary of Safety Consequence/Risk Band IV

• S.4/CSS-to-Core Barrel Bolts (bounding)/SCC **[Cascading]**
• B.4/Lower Grid Assembly-to-Core Barrel Bolts (bounding)/SCC **[Cascading]**
• L.2.3/Flow Distributor-to-Shell Forging Bolts (bounding)/SCC **[Cascading]**

Table 5-3
Summary of Safety Consequences/Risk Band III

- P.1.2/Weldment Ribs/Wear
- P.1.4/Support Flange/Wear
- S.2/CSS Top Flange/Wear
- B.17/Baffle Plate/IASCC
- B.18/Former Plate/IASCC
- B.18/Former Plate/Void Swelling
- B.19/Core Barrel-to-Former Plate Cap Screws/IASCC
- B.19/Core Barrel-to-Former Plate Cap Screws/Fatigue (T&ISR/C)
- B.19/Core Barrel-to-Former Plate Cap Screws/Wear (T&ISR/C)
- B.22/Baffle Plate-to-Former Plate Bolts/IASCC
- B.22/Baffle Plate-to-Former Plate Bolts/Fatigue (T&ISR/C)
- B.22/Baffle Plate-to-Former Plate Bolts/Wear (T&ISR/C)
- B.24/Baffle Plate-to-Former Plate Shoulder Screws/IASCC
- B.24/Baffle Plate-to-Former Plate Shoulder Screws/Fatigue (T&ISR/C)
- B.24/Baffle Plate-to-Former Plate Shoulder Screws/Wear (T&ISR/C)
- B.26/Baffle Plate-to-Baffle Plate Bolts/IASCC
- B.26/Baffle Plate-to-Baffle Plate Bolts/Fatigue (T&ISR/C)
- B.26/Baffle Plate-to-Baffle Plate Bolts/Wear (T&ISR/C)
- S.4/CSS-to-Core Barrel Bolts (bounding)/Fatigue (TSR/C) **[Cascading]**
- S.4/CSS-to-Core Barrel Bolts (bounding)/Wear (TSR/C) **[Cascading]**
- B.4/Lower Grid Assembly-to-Core Barrel Bolts (bounding)/Fatigue (TSR/C) **[Cascading]**
- B.4/Lower Grid Assembly-to-Core Barrel Bolts (bounding)/Wear (TSR/C) **[Cascading]**
- L.2.3/Flow Distributor-to-Shell Forging Bolts (bounding)/Fatigue (OCL & TSR/C) **[Cascading]**
- L.2.3/Flow Distributor-to-Shell Forging Bolts (bounding)/Wear (TSR/C) **[Cascading]**

Tables 5-4 through 5-6 summarize the individual contributors for the economic risk for the highest three risk bands. Table 5-4 shows the same cascading failures as in the safety consequence Risk Band IV. Table 5-5 shows the same wear failures as in the safety consequence Risk Band III. The economic risk in Risk Band III (moderate risk) pushes further in the FMECA than the highest three risk bands of safety consequence and identifies a number of component item/mechanism combinations that do not appear in the safety consequence risk lists (Table 5-6).

Table 5-4
Summary of Economic Consequences/Risk Band V

- S.4/CSS-to-Core Barrel Bolts (bounding)/SCC **[Cascading]**
- B.4/Lower Grid Assembly-to-Core Barrel Bolts (bounding)/SCC **[Cascading]**
- L.2.3/Flow Distributor-to-Shell Forging Bolts (bounding)/SCC **[Cascading]**

Table 5-5
Summary of Economic Consequences/Risk Band IV

- P.1.2/Weldment Ribs/Wear
- P.1.4/Support Flange/Wear
- S.2/CSS Top Flange/Wear
- S.4/CSS-to-Core Barrel Bolts (bounding)/Fatigue (TSR/C) **[Cascading]**
- S.4/CSS-to-Core Barrel Bolts (bounding)/Wear (TSR/C) **[Cascading]**
- B.4/Lower Grid Assembly-to-Core Barrel Bolts (bounding)/Fatigue (TSR/C) **[Cascading]**
- B.4/Lower Grid Assembly-to-Core Barrel Bolts (bounding)/Wear (TSR/C) **[Cascading]**
- L.2.3/Flow Distributor-to-Shell Forging Bolts (bounding)/Fatigue (OCL & TSR/C) **[Cascading]**
- L.2.3/Flow Distributor-to-Shell Forging Bolts (bounding)/Wear (TSR/C) **[Cascading]**

Table 5-6
Summary of Economic Consequence/Risk Band III

- P.1.8/Base Blocks/SCC
- P.1.9/Lifting Lugs-to-Base Block Bolts/Fatigue (TSR/C)
- P.1.9/Lifting Lugs-to-Base Block Bolts/Wear (TSR/C)
- P.1.11/Integral Lifting Lug/Base Block (ONS-1)/SCC
- P.3.7/Cap Screws (Support Pads)/Fatigue (T&ISR/C)
- P.3.7/Cap Screws (Support Pads)/Wear (T&ISR/C)
- P.4.5/CRGT Spacer Castings/Thermal Aging Embrittlement
- P.4.8/CRGT Rod Guide Tubes/Wear
- P.4.9/CRGT Rod Guide Sectors/Wear
- S.1/CSS Cylinder/Fatigue
- S.4/CSS-to-Core Barrel Bolts (bounding)/SCC
- S.6/Outlet Nozzles (ONS-3) (bounding)/Thermal Aging Embrittlement
- S.10/Vent Valve Retaining Rings/Thermal Aging Embrittlement
- S.12/Vent Valve Disc/Thermal Aging Embrittlement
- S.14/Vent Valve Disc Shaft/Hinge Pin/Thermal Aging Embrittlement
- S.17/CSS Lifting Lugs/SCC
- B.1/Core Barrel Cylinder/Irradiation Embrittlement
- B.4/Lower Grid Assembly-to-Core Barrel Bolts (bounding)/SCC
- B.6/(Upper) Thermal Shield-to-Core Barrel Bolts (bounding)/SCC
- B.8/Replacement Surveillance Specimen Holder Tube-to-Thermal Shield Studs/Nuts/Bolts (bounding)/SCC
- B.10/Thermal Shield Cylinder/Wear

Table 5-6 (continued)
Summary of Economic Consequence/Risk Band III

- B.16/Thermal Shield Cap Screw/Fatigue (T&ISR/C)
- B.16/Thermal Shield Cap Screw/Wear (T&ISR/C)
- B.17/Baffle Plates/IASCC
- B.17/Baffle Plates/Irradiation Embrittlement
- B.18/Former Plates/IASCC
- B.18/Former Plates/Irradiation Embrittlement

- B.18/Former Plates/Void Swelling
- B.19/Core Barrel-to-Former Plate Cap Screws/IASCC
- B.19/Core Barrel-to-Former Plate Cap Screws/Fatigue (T&ISR/C)
- B.19/Core Barrel-to-Former Plate Cap Screws/Irradiation Embrittlement
- B.19/Core Barrel-to-Former Plate Cap Screws/Wear (T&ISR/C)
- B.22/Baffle Plate-to-Former Plate Bolts/IASCC
- B.22/Baffle Plate-to-Former Plate Bolts/Fatigue (T&ISR/C)
- B.22/Baffle Plate-to-Former Plate Bolts/Irradiation Embrittlement
- B.22/Baffle Plate-to-Former Plate Bolts/Wear (T&ISR/C)
- B.24/Baffle Plate-to-Former Plate Shoulder Screws/IASCC
- B.24/Baffle Plate-to-Former Plate Shoulder Screws/Fatigue (T&ISR/C)
- B.24/Baffle Plate-to-Former Plate Shoulder Screws/Irradiation Embrittlement
- B.24/Baffle Plate-to-Former Plate Shoulder Screws/Wear (T&ISR/C)
- B.26/Baffle Plate-to-Baffle Plate Bolts/IASCC
- B.26/Baffle Plate-to-Baffle Plate Bolts/Fatigue (T&ISR/C)
- B.26/Baffle Plate-to-Baffle Plate Bolts/Irradiation Embrittlement
- B.26/Baffle Plate-to-Baffle Plate Bolts/Wear (T&ISR/C)
- L.1.4/Cap Screws/Fatigue (T&ISR/C)
- L.1.4/Cap Screws/Wear (T&ISR/C)
- L.1.5/Rib Section-to-Shell Forging Cap Screws/Fatigue (TSR/C)
- L.1.5/Rib Section-to-Shell Forging Cap Screws/Irradiation Embrittlement
- L.1.5/Rib Section-to-Shell Forging Cap Screws/Wear (TSR/C)
- L.1.11/Lower Grid Assembly-to-Thermal Shield Bolts (bounding)/SCC
- L.1.17/Shock Pad Bolts/SCC
- L.2.3/Flow Distributor-to-Shell Forging Bolts/SCC

In addition to the component items/degradation mechanism reflected above, there are a number of combinations that while identified as improbable will either result in severe consequences, affect the ability to cope with a LOCA, or will require the successful "operation" of the guide lugs. Accordingly, while not classified into a specific risk band, these component items, potentially susceptible to CCF, should continue to fall under current ASME Section XI visual examinations (VT-3); these are provided in Table 5-7.

Table 5-7
Summary of "Improbable" Component Item/Degradation Mechanism Combinations

- P.2.5/Round Bars/SCC

- S.2/CSS Top Flange/SCC
- S.3/CSS Bottom Flange/SCC
- S.15/CSS Round Bars/SCC
- B.2/Core Barrel Assembly Top Flange/SCC
- B.3/Core Barrel Assembly Bottom Flange/SCC
- L.1.17/Shock Pad Bolts/SCC
- L.1.17/Shock Pad Bolts/Fatigue (TSR/C)
- L.1.17/Shock Pad Bolts/Wear (TSR/C)
- L.2.1/Flow Distributor Head/SCC
- L.2.2/Flow Distributor Flange/SCC

While not listed specifically in any of the results summary tables above, there is one additional set of component items whose importance should not be overlooked – guide lugs. These component items do not appear in the risk matrix since the guide lugs are not within the scope of the PWR internals, but rather are considered part of the reactor vessel. However, the importance of the guide lugs is highlighted in the FMECA with bold text in the comments column. The successful operation of the guide lugs are essential to safety in the event that the PWR internals fail due to one or more failure modes identified in the FMECA. Accordingly, the reactor vessel guide lugs should be considered as part of whatever inspection strategy is developed for the internals.

6
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[3] Materials Reliability Program: PWR Internals Material Aging Degradation Mechanism Screening and Threshold Values (MRP-175). EPRI, Palo Alto, CA: 2005. 1012081.

[4] NEI Document NEING0620033926, "NEI 03-08, Guideline for the Management of Materials Issues," May 2003.

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A
FAILURE MODES, EFFECTS, AND CRITICALITY ANALYSIS (FMECA) TABLE

Appendix A contains the four FMECA tables that comprise the entirety of this analysis. The tables are divided by PWR internals assemblies as follows:

- Table A-1 Plenum Assembly
- Table A-2 Core Support Shield Assembly
- Table A-3 Core Barrel Assembly
- Table A-4 Lower Internals Assembly

Table A-1
Plenum Assembly (P)

Component Item Name	FMECA Identifier	Degradation Mechanism	Failure Mode	Failure Effect (Local)	Failure Effect (Global)	Susceptibility	Severity of Consequences (Safety)	Severity of Consequences (Economic)	Detectable	Comments
Plenum Cover Assembly	P.1	Function	Provides support for the top of the 69 control rod guide tube (CRGT) assemblies.							
Weldment Ribs	P.1.1	SCC	Cracking/ Fracture	Localized degradation of structural integrity (at the heat-affected zone, HAZ)	No operational effects. With cracking, ribs would be in place to provide support for the control rod guide tube (CRGT) assemblies.	B	1	2	Portions of the weldment ribs are accessible for a VT-3 (gross cracking) or enhanced VT-1 or UT during refueling or 10-year ISI.	No alignment issues as CRGTs are welded to the cover plate. While there are no safety consequences, economic consequences incurred due to discovery could be either analysis or repair (grinding).
Weldment Rib Pads	P.1.2	SCC	Cracking/ Fracture	Loss of structural integrity (at the HAZ)	No operational effects.	B	1	2	Pads are accessible for a VT-3 inspection (gross cracking) during refueling or 10-year ISI.	
		Wear	Loss of material	Affect mating surface to the reactor vessel head	Loss of clamp-up, which could lead to internals motion (rocking) that could result in fuel failure.	C	2	3	Yes. Possibly with radiation monitoring, though could not pin-point the cause.	Reactor vessel head sits on top of the rib pads. Economic consequences driven by first of a kind difficult repair.
Bottom Flange	P.1.3	SCC	Cracking/ Fracture	Localized degradation of structural integrity (at the HAZ)	No operational effects.	B	1	2	No. Could be discovered with UT during 10-year ISI.	Economic consequences driven by analysis.
					Critical flaw size (crack) around the weld separating the plenum cover assembly from the plenum cylinder.	B/I	Not evaluated	Not evaluated		
Support Flange	P.1.4	SCC	Cracking/ Fracture	Localized degradation of structural integrity (at the HAZ)	No operational effects. Cracking could affect subsequent alignment when placing the plenum assembly into the core support shield assembly.	B	1	2	No.	Economic consequences driven by analysis.
		Wear	Loss of material	Affects mating surface to the reactor vessel head	Loss of clamp-up, which could lead to internals motion (rocking) that could result in fuel failure.	C	2	3	Yes. Possibly with radiation monitoring, though could not pin-point the cause.	Economic consequences driven by first of a kind difficult repair.

Failure Modes, Effects, and Criticality Analysis (FMECA) Table

Component Item Name	FMECA Identifier	Degradation Mechanism	Failure Mode	Failure Effect (Local)	Failure Effect (Global)	Susceptibility	Severity of Consequences (Safety)	Severity of Consequences (Economic)	Detectable	Comments
Support Ring	P.1.5	SCC	Cracking/ Fracture	Localized degradation of structural integrity (at the HAZ)	No operational effects. Cracking could affect mating alignment with the reactor vessel head.	B	1	2	No.	Economic consequences driven by analysis.
Cover Plate	P.1.6	SCC	Cracking/ Fracture	Localized degradation of structural integrity (at the HAZ)	No operational effects.	B	1	2	No.	Economic consequences driven by analysis.
Lifting Lugs	P.1.7	No credible degradation mechanism	n/a	n/a	n/a	A	Not evaluated	Not evaluated		The screening analysis shows that none of the screening parameter values are exceeded (Category A) and therefore the likelihood of any consequential age-related degradation is extremely small. Accordingly, this component item is only evaluated (and screened) on a susceptibility basis. (No evaluation of consequence is performed.)
Base Blocks	P.1.8	SCC	Cracking/ Fracture	Localized degradation of structural integrity (at the HAZ)	No operational effects. Base blocks will not hold when plenum assembly is removed during refueling.	B	1	3	Yes. If failed during refueling.	Inability to remove Plenum Assembly could significantly delay a refueling outage while repairs were being performed. If weld(s) fail while Plenum Assembly is being lifted, the resulting fall could damage other internals component items. With such small clearances, the plenum assembly would be likely to fall straight down.
Lifting Lug-to-Base Block Bolts	P.1.9	Fatigue (Thermal Stress Relaxation/ Creep)	Cracking/ Fracture	Loss of mechanical joint integrity	No operational effects. If locking cups fail (P.1.10 has no credible degradation mechanism), broken pieces from locking cups and/or bolts could result in loose parts. Also applies to wear.	B	1	3	Yes. If failed during refueling.	Inability to remove Plenum Assembly could significantly delay a refueling outage while repairs were being performed. If bolts(s) fail while Plenum Assembly is being lifted, the resulting fall could damage other internals component items. With such small clearances, the plenum assembly would be likely to fall straight down. While loose parts are possible, this is considered part of normal operation.
		Wear (Thermal Stress Relaxation/Creep)	Loss of material	Loss of mechanical joint integrity	No operational effects.	B	1	3	Yes. If failed during refueling.	Inability to remove Plenum Assembly could significantly delay a refueling outage while repairs were being performed. If bolt(s) fail while Plenum Assembly is being lifted, the resulting fall could damage other internals component items. With such small clearances, the plenum assembly would be likely to fall straight down. While loose parts are possible, this is considered part of normal operation.
Locking Cups	P.1.10	No credible degradation mechanism	n/a	n/a	n/a	A	Not evaluated	Not evaluated		The screening analysis shows that none of the screening parameter values are exceeded (Category A) and therefore the likelihood of any consequential age-related degradation is extremely small. Accordingly, this component item is only evaluated (and screened) on a susceptibility basis. (No evaluation of consequence is performed.)

Component Item Name	FMECA Identifier	Degradation Mechanism	Failure Mode	Failure Effect (Local)	Failure Effect (Global)	Susceptibility	Severity of Consequences (Safety)	Severity of Consequences (Economic)	Detectable	Comments
Integral Lifting Lugs/ Base Blocks (ONS-1)	P.1.11	SCC	Cracking/ Fracture	Localized degradation of structural integrity (at the HAZ)	No operational effects. Base blocks will not hold when plenum assembly is removed during refueling.	B	1	3	Yes. If failed during refueling.	Inability to remove Plenum Assembly could significantly delay a refueling outage while repairs were being performed. If weld(s) fail while Plenum Assembly is being lifted, the resulting fall could damage other internals component items. With such small clearances, the plenum assembly would be likely to fall straight down.
Plenum Cover Letter Plate	P.1.12	SCC	Cracking/ Fracture	Localized degradation of structural integrity (at the HAZ)	No operation effects.	B	1	1	No.	
Plenum Cylinder Assembly	P.2	Function	Directs the flow of reactor coolant from the core area to the reactor vessel outlet nozzles.							
Cylinder	P.2.1	SCC	Cracking/ Fracture	Localized degradation of structural integrity (at the HAZ)	No operational effects.	B	1	2	No.	SCC cracking is tight. Does not permit any significant flow diversion. SCC cracking potential is at the heat affected zone (HAZ) and the flange welds (see items P.2.2 and P.2.3). Economic consequences driven by analysis.
		Fatigue	Cracking/ Fracture	Localized degradation of structural integrity; flow diversion	No operational effects.	A	Not evaluated	Not evaluated		Screening parameter (estimated CUF value) may be incorrect.
Top Flange	P.2.2	SCC	Cracking/ Fracture	Localized degradation of structural integrity (at the HAZ)	No operational effects.	B	1	2	No. Could be discovered with enhanced VT-1 or UT during 10-year ISI.	Economic consequences driven by analysis.
					Critical flaw size (crack) around the weld separating the plenum cover assembly from the plenum cylinder.	B/I	Not evaluated	Not evaluated		
Bottom Flange	P.2.3	SCC	Cracking/ Fracture	Localized degradation of structural integrity (at the HAZ)	No operational effects.	B	1	2	No. Could be discovered with enhanced VT-1 or UT during 10-year ISI.	Economic consequences driven by analysis.
					Critical flaw size (crack) around the weld separating the plenum cylinder from the upper grid assembly.	B/I	Not evaluated	Not evaluated		
Reinforcing Plates	P.2.4	SCC	Cracking/ Fracture	Localized degradation of structural integrity (at the HAZ)	No operational effects.	B	1	2	No.	Economic consequences driven by analysis.
Round Bars	P.2.5	SCC	Cracking/ Fracture	Localized degradation of structural integrity (at the HAZ)	Loose parts.	B/I	Not evaluated	Not evaluated	Loose Parts Monitoring System (LPMS)	In the event of a hot leg LOCA, the round bars will prevent the plenum assembly and the core support shield assembly from collapsing.
					Loss of integrity of most or all of the round bars in both the Plenum Cylinder Assembly and the Core Support Shield Assembly could restrict coolant flow during a response to a LOCA. See item S.14.	**B/I**	**Not evaluated**	**Not evaluated**		

Failure Modes, Effects, and Criticality Analysis (FMECA) Table

Component Item Name	FMECA Identifier	Degradation Mechanism	Failure Mode	Failure Effect (Local)	Failure Effect (Global)	Susceptibility	Severity of Consequences (Safety)	Severity of Consequences (Economic)	Detectable	Comments
Top Flange-to-Cover Bolts	P.2.6	Fatigue (Thermal Stress Relaxation/ Creep)	Cracking/ Fracture	Loss of mechanical joint integrity	No operational effects if a small number of bolts are degraded. If locking cups fail (P.2.7 has no credible degradation mechanism), broken pieces from locking cups and/or bolts could result in loose parts. Also applies to wear. If a significant number of bolts were degraded, the plenum cover might separate from the cylinder, however the welded and bolted control rod guide tubes will probably hold the plenum assembly together.	B B/I	1 Not evaluated	2 Not evaluated	No.	Economic consequences driven by analysis. A large number of bolts failing is not a cascading (or dependent) failure since these bolts do not bear load; when (and if) one bolt breaks, load is not then distributed to the other bolts. Therefore, it is improbable that a large number of these bolts will fail simultaneously. While loose parts are possible, this is considered part of normal operation.
		Wear (Thermal Stress Relaxation/Creep)	Loss of material	Loss of mechanical joint integrity	No operational effects if a small number of bolts are degraded. If a significant number of bolts were degraded, the plenum cover might separate from the cylinder, however the welded and bolted control rod guide tubes will probably hold the plenum assembly together.	B B/I	1 Not evaluated	2 Not evaluated	No.	Economic consequences driven by analysis. A large number of bolts failing is not a cascading (or dependent) failure since these bolts do not bear load; when (and if) one bolt breaks, load is not then distributed to the other bolts. Therefore, it is improbable that a large number of these bolts will fail simultaneously. While loose parts are possible, this is considered part of normal operation.
Locking Cups	P.2.7	No credible degradation mechanism	n/a	n/a	n/a	A	Not evaluated	Not evaluated		The screening analysis shows that none of the screening parameter values are exceeded (Category A) and therefore the likelihood of any consequential age-related degradation is extremely small. Accordingly, this component item is only evaluated (and screened) on a susceptibility basis. (No evaluation of consequence is performed.)

Component Item Name	FMECA Identifier	Degradation Mechanism	Failure Mode	Failure Effect (Local)	Failure Effect (Global)	Susceptibility	Severity of Consequences (Safety)	Severity of Consequences (Economic)	Detectable	Comments
Bottom Flange-to-Upper Grid Assembly Bolts	P.2.8	Fatigue (Operational Cyclic Loading and Thermal Stress Relaxation/Creep)	Cracking/ Fracture	Loss of mechanical joint integrity	No operational effects if a small number of bolts are degraded. If locking cups fail (P.2.9 has no credible degradation mechanism), broken pieces from locking cups and/or bolts could result in loose parts. Also applies to wear. If a significant number of bolts were degraded, the plenum cylinder might separate from the upper grid assembly, however the welded and bolted control rod guide tubes will probably hold the plenum assembly together.	B B/I	1 Not evaluated	2 Not evaluated	No.	Economic consequences driven by analysis. A large number of bolts failing is not a cascading (or dependent) failure since these bolts do not bear load; when (and if) one bolt breaks, load is not then distributed to the other bolts. Therefore, it is improbable that a large number of these bolts will fail simultaneously. While loose parts are possible, this is considered part of normal operation.
		Wear (Thermal Stress Relaxation/Creep)	Loss of material	Loss of mechanical joint integrity	No operational effects if a small number of bolts are degraded. If a significant number of bolts were degraded, the plenum cylinder might separate from the upper grid assembly, however the welded and bolted control rod guide tubes will probably hold the plenum assembly together.	B B/I	1 Not evaluated	2 Not evaluated	No.	Economic consequences driven by analysis. A large number of bolts failing is not a cascading (or dependent) failure since these bolts do not bear load; when (and if) one bolt breaks, load is not then distributed to the other bolts. Therefore, it is improbable that a large number of these bolts will fail simultaneously. While loose parts are possible, this is considered part of normal operation.
Locking Cups	P.2.9	No credible degradation mechanism	n/a	n/a	n/a	A	Not evaluated	Not evaluated		The screening analysis shows that none of the screening parameter values are exceeded (Category A) and therefore the likelihood of any consequential age-related degradation is extremely small. Accordingly, this component item is only evaluated (and screened) on a susceptibility basis. (No evaluation of consequence is performed.)
Upper Grid Assembly	P.3	Function	Provides support and seating surface for the tops of the fuel assemblies (located in the core barrel below). Provides restraint and alignment for the bottoms of the CRGT assemblies.							

Failure Modes, Effects, and Criticality Analysis (FMECA) Table

Component Item Name	FMECA Identifier	Degradation Mechanism	Failure Mode	Failure Effect (Local)	Failure Effect (Global)	Susceptibility	Severity of Consequences (Safety)	Severity of Consequences (Economic)	Detectable	Comments
Rib Section	P.3.1	Fatigue	Cracking/ Fracture	Localized degradation of structural integrity	No operational effects. Ribs would continue to provide alignment for in-place fuel assemblies. May impact refueling when fuel assemblies are removed. Reloading may be affected by failed rib sections.	A	Not evaluated	Not evaluated		Screening parameter (estimated CUF value) may be incorrect.
Ring Forging	P.3.2	No credible degradation mechanism	n/a	n/a	n/a	A	Not evaluated	Not evaluated		The screening analysis shows that none of the screening parameter values are exceeded (Category A) and therefore the likelihood of any consequential age-related degradation is extremely small. Accordingly, this component item is only evaluated (and screened) on a susceptibility basis. (No evaluation of consequence is performed.)
Rib-to-Ring Cap Screws	P.3.3	Fatigue (Thermal Stress Relaxation/ Creep)	Cracking/ Fracture	Loss of mechanical joint integrity	No operational effects. If locking pins fail (P.3.4 has no credible degradation mechanism), broken pieces from locking pins and/or screws could result in loose parts. Also applies to wear.	B	1	2	No.	Economic consequences driven by delayed outage when repairs are made. While loose parts are possible, this is considered part of normal operation.
		Wear (Thermal Stress Relaxation/Creep)	Loss of material	Loss of mechanical joint integrity	No operational effects.	B	1	2	No.	Economic consequences driven by delayed outage when repairs are made. While loose parts are possible, this is considered part of normal operation.
Locking Pins	P.3.4	No credible degradation mechanism	n/a	n/a	n/a	A	Not evaluated	Not evaluated		The screening analysis shows that none of the screening parameter values are exceeded (Category A) and therefore the likelihood of any consequential age-related degradation is extremely small. Accordingly, this component item is only evaluated (and screened) on a susceptibility basis. (No evaluation of consequence is performed.)
Fuel Assembly Support Pads	P.3.5	Irradiation Embrittlement	Less flaw tolerant	More susceptible to crack growth when subjected to an external load with a flaw	No operational effects.	B	1	2	No.	More susceptible to crack growth from an external load/force. Decrease in toughness. Economic consequences driven by analysis and/or repairs.
Dowels	P.3.6	Irradiation Embrittlement	Less flaw tolerant	More susceptible to crack growth when subjected to an external load with a flaw	No operational effects.	B	1	2	No.	More susceptible to crack growth from an external load/force. Decrease in toughness. Economic consequences driven by analysis and/or repairs.

Component Item Name	FMECA Identifier	Degradation Mechanism	Failure Mode	Failure Effect (Local)	Failure Effect (Global)	Susceptibility	Severity of Consequences (Safety)	Severity of Consequences (Economic)	Detectable	Comments
Cap Screws	P.3.7	Fatigue (Thermal and Irradiation Stress Relaxation/Creep)	Cracking/ Fracture	Loss of mechanical joint integrity	No operational effects.	C	1	2	No.	The locking "mechanism" for these cap screws is a fillet weld. Economic consequences driven by analysis and/or repairs.
		Wear (Thermal and Irradiation Stress Relaxation/Creep)	Loss of material	Loss of mechanical joint integrity	No operational effects.	C	1	2	No.	Economic consequences driven by analysis and/or repairs. Wear, when driven by irradiation stress relaxation/creep, has a susceptibility of "C." Also, note, that wear, when driven just by thermal stress relaxation/creep, has a susceptibility of "B."
		Irradiation Embrittlement	Less flaw tolerant	More susceptible to crack growth when subjected to an external load with a flaw	No operational effects.	B	1	2	No.	More susceptible to crack growth from an external load/force. Decrease in toughness. Economic consequences driven by analysis and/or repairs.
Control Rod Guide Tube Assembly	P.4	Function	Provides control rod assemblies (CRAs) guidance, protects the CRA (from the effects of coolant cross-flow), and structurally connects the upper grid assembly to the plenum cover							
Pipe	P.4.1	SCC	Cracking/ Fracture	Localized degradation of structural integrity (at the HAZ)	No operational effects. Critical flaw size (crack) around the weld separating the CRGT pipe from the flange.	B B/I	1 Not evaluated	2 Not evaluated	Yes. Control rod drive testing before and after refueling will un-cover any degrad-ation in CR insertion.	SCC cracking is tight. Does not permit any significant flow diversion. Higher than normal susceptibility due to fluoride contamination in the partial penetration weld. Economic consequences driven by analysis and/or repair.
Flange	P.4.2	SCC	Cracking/ Fracture	Localized degradation of structural integrity (at the HAZ)	No operational effects. Critical flaw size (crack) around the weld separating the CRGT pipe from the flange.	B B/I	1 Not evaluated	2 Not evaluated	Yes. Control rod drive testing before and after refueling will un-cover any degrad-ation in CR insertion.	SCC cracking is tight. Does not permit any significant flow diversion. Higher than normal susceptibility due to fluoride contamination in the partial penetration weld. Economic consequences driven by analysis and/or repair.
Flange-to-Upper Grid Cap Screws	P.4.3	Fatigue (Thermal Stress Relaxation/ Creep)	Cracking/ Fracture	Loss of mechanical joint integrity	No operational effects. If fillet weld fails (no credible degradation mechanism identified), broken pieces from weld and/or bolts could result in loose parts. Also applies to wear.	B	1	2	Yes. Control rod drive testing before and after refueling will un-cover any degrad-ation in CR insertion.	The locking "mechanism" for these bolts is a fillet weld.
		Wear (Thermal Stress Relaxation/Creep)	Loss of material	Loss of mechanical joint integrity	No operational effects.	B	1	2	Yes. Control rod drive testing before and after refueling will un-cover any degrad-ation in CR insertion.	The locking "mechanism" for these bolts is a fillet weld.
Dowels	P.4.4	No credible degradation mechanism	n/a	n/a	n/a	A	Not evaluated	Not evaluated		The screening analysis shows that none of the screening parameter values are exceeded (Category A) and therefore the likelihood of any consequential age-related degradation is extremely small. Accordingly, this component item is only evaluated (and screened) on a susceptibility basis. (No evaluation of consequence is performed.)

Failure Modes, Effects, and Criticality Analysis (FMECA) Table

Component Item Name	FMECA Identifier	Degradation Mechanism	Failure Mode	Failure Effect (Local)	Failure Effect (Global)	Susceptibility	Severity of Consequences (Safety)	Severity of Consequences (Economic)	Detectable	Comments
Spacer Castings	P.4.5	Thermal Aging Embrittlement	Less flaw tolerant	More susceptible to crack growth when subjected to an external load with a flaw	No operational effects.	C	1	2	Yes. Control rod drive testing before and after refueling will un-cover any degradation in CR insertion.	More susceptible to crack growth from an external load/force. Decrease in toughness.
Spacer Castings Cap Screws	P.4.6	Fatigue (Thermal Stress Relaxation/ Creep)	Cracking/ Fracture	Loss of mechanical joint integrity	No operational effects. Broken pieces from screws could result in loose parts. Also applies to wear.	B	1	2	Yes. Control rod drive testing before and after refueling will un-cover any degradation in CR insertion.	
		Wear (Thermal Stress Relaxation/Creep)	Loss of material	Loss of mechanical joint integrity	No operational effects.	B	1	2	Yes. Control rod drive testing before and after refueling will un-cover any degradation in CR insertion.	
Spacer Castings Washers	P.4.7	No credible degradation mechanism	n/a	n/a	n/a	A	Not evaluated	Not evaluated		The screening analysis shows that none of the screening parameter values are exceeded (Category A) and therefore the likelihood of any consequential age-related degradation is extremely small. Accordingly, this component item is only evaluated (and screened) on a susceptibility basis. (No evaluation of consequence is performed.)
Rod Guide Tubes	P.4.8	Wear	Loss of material	Wear to enlarge the opening in the rod guide tubes.	Control rod "pops" out of the enlarged opening and gets stuck.	B	2	3	Yes. Control rod drive testing before and after refueling will un-cover any degradation in CR insertion.	Economic consequences driven by extended refueling outage and control rod repair and testing.
Rod Guide Sectors	P.4.9	Wear	Loss of material	Wear to enlarge the opening in the rod guide sectors.	Control rod "pops" out of the enlarged opening and gets stuck.	B	2	3	Yes. Control rod drive testing before and after refueling will un-cover any degradation in CR insertion.	Economic consequences driven by extended refueling outage and control rod repair and testing.

Table A-2
Core Support Shield Assembly (S)

Component Item Name	FMECA Identifier	Degradation Mechanism	Failure Mode	Failure Effect (Local)	Failure Effect (Global)	Susceptibility	Severity of Consequences (Safety)	Severity of Consequences (Economic)	Detectable	Comments
Core Support Shield Assembly (S)	S	Function	Provides a boundary between the incoming cold reactor coolant (CSS outside) and heated reactor coolant (CSS inside) Supports all reactor vessel internals (top flange); specifically supports the plenum assembly CSS includes vent valves that vent steam in the event of a break of a reactor vessel inlet pipe							
Cylinder	S.1	SCC	Cracking/ Fracture	Localized degradation of structural integrity (at the HAZ)	No operational effects.	B	1	2	No.	SCC cracking is tight. Does not permit any significant flow diversion. SCC cracking potential is at the heat affected zone (HAZ) and the flange welds (see items S.2 and S.3).
		Fatigue	Cracking/ Fracture	Localized degradation of structural integrity; leakage	Increased bypass flow, possibly leading to critical heat flux (CHF) conditions.	B	1	3	No. (See comments)	For a large crack (improbable), increase in temperature would be measured by the core exit thermocouples.
Top Flange	S.2	SCC	Cracking/ Fracture	Localized degradation of structural integrity (at the HAZ)	No operational effects. Critical flaw size (crack) around the weld separating the top flange from the core support shield assembly cylinder (causing the internals and core to fall and increase bypass flow).	B **B/I**	1 **Not evaluated**	2 **Not evaluated**	No.	Reactor vessel guide lugs are designed to halt the fall of the internals/core (onto the shock pads) and not affect control rod insertion. However, note, that while not in the PWR internals scope, the guide lugs need to maintain functionality to stop the fall of the internals.
		Wear	Loss of material	None	Loss of clamp-up, which could lead to internals motion (rocking) that could result in fuel failure.	C	2	3	Yes. Possibly with radiation monitoring, though could not pin-point the cause.	CSS top flange sits on top of the reactor vessel flange and supports all of the internals. Economic consequences driven by first of a kind difficult repair.
Bottom Flange	S.3	SCC	Cracking/ Fracture	Localized degradation of structural integrity (at the HAZ)	No operational effects. Critical flaw size (crack) around the weld separating the core support shield assembly cylinder from the bottom flange (causing the internals and core to fall and increase bypass flow).	B **B/I**	1 **Not evaluated**	2 **Not evaluated**	No.	Reactor vessel guide lugs are designed to halt the fall of the internals/core (onto the shock pads) and not affect control rod insertion. However, note, that while not in the PWR internals scope, the guide lugs need to maintain functionality to stop the fall of the internals.
Original CSS-to-Core Barrel Bolts (ANO-1,	S.4a	SCC	Cracking/ Fracture	Localized degradation of structural integrity	No effect if a small number of bolts are degraded.	D	1	2	No.	These bolts subject to SCC due to the material type and stresses. There are 120 CSS-to-core barrel bolts, which represents a

Failure Modes, Effects, and Criticality Analysis (FMECA) Table

Component Item Name	FMECA Identifier	Degradation Mechanism	Failure Mode	Failure Effect (Local)	Failure Effect (Global)	Susceptibility	Severity of Consequences (Safety)	Severity of Consequences (Economic)	Detectable	Comments
DB, ONS)					If a significant number of bolts were degraded, the internals (core barrel assembly, core, and lower internals assembly) would drop (and increase bypass flow).	D	3*	4*	Yes. A number of detection means are possible: neutron noises, radioactivity alarms, loose parts monitoring.	significant level of redundancy, i.e., a large number of bolts would have to fail before the internals would drop. The asterisks (*) indicate that the consequences are as a result of a cascading (or dependent) failure. As some bolts fail, the remaining bolts will be subject to a greater load, which will increase their likelihood of failure. (Only applies to ONS, since ANO-1 has replaced all but six bolts, and Davis-Besse all but three bolts.) Reactor vessel guide lugs are designed to halt the fall of the internals/core (onto the shock pads) and not affect control rod insertion. However, note, that while not in the PWR internals scope, the guide lugs need to maintain functionality to stop the fall of the internals.
		Fatigue (Thermal Stress Relaxation/ Creep)	Cracking/ Fracture	Loss of mechanical joint integrity	No effect if a small number of bolts are degraded. If locking clips fail (S.5 has no credible degradation mechanism), broken pieces from locking clips and/or bolts could result in loose parts. Also applies to wear.	B	1	2	No.	There are 120 CSS-to-core barrel bolts, which represents a significant level of redundancy, i.e., a large number of bolts would have to fail before the internals would drop. The asterisks (*) indicate that the consequences are as a result of a cascading (or dependent) failure. As some bolts fail, the remaining bolts will be subject to a greater load, which will increase their likelihood of failure. (Only applies to ONS, since ANO-1 has replaced all but six bolts, and Davis-Besse all but three bolts.) Reactor vessel guide lugs are designed to halt the fall of the internals/core (onto the shock pads) and not affect control rod insertion. However, note, that while not in the PWR internals scope, the guide lugs need to maintain functionality to stop the fall of the internals.
					If a significant number of bolts were degraded, the internals (core barrel assembly, core, and lower internals assembly) would drop (and increase bypass flow).	B	3*	4*	Yes. A number of detection means are possible: neutron noises, radioactivity alarms, loose parts monitoring.	
		Wear (Thermal Stress Relaxation/Creep)	Loss of material	Loss of mechanical joint integrity	No effect if a small number of bolts are degraded.	B	1	2	No.	There are 120 CSS-to-core barrel bolts, which represents a significant level of redundancy, i.e., a large number of bolts would have to fail before the internals would drop. The asterisks (*) indicate that the consequences are as a result of a cascading (or dependent) failure. As some bolts fail, the remaining bolts will be subject to a greater load, which will increase their likelihood of failure. (Only applies to ONS, since ANO-1 has replaced all but six bolts, and Davis-Besse all but three bolts.) Reactor vessel guide lugs are designed to halt the fall of the internals/core (onto the shock pads) and not affect control rod insertion. However, note, that while not in the PWR internals scope, the guide lugs need to maintain functionality to stop the fall of the internals.
					If a significant number of bolts were degraded, the internals (core barrel assembly, core, and lower internals assembly) would drop (and increase bypass flow).	B	3*	4*	Yes. A number of detection means are possible: neutron noises, radioactivity alarms, loose parts monitoring.	

Component Item Name	FMECA Identifier	Degradation Mechanism	Failure Mode	Failure Effect (Local)	Failure Effect (Global)	Susceptibility	Severity of Consequences (Safety)	Severity of Consequences (Economic)	Detectable	Comments
Original CSS-to-Core Barrel Bolts (TMI-1)	S.4b	SCC	Cracking/ Fracture	Localized degradation of structural integrity	No effect if a small number of bolts are degraded.	D	1	2	No.	These bolts subject to SCC due to the material type and stresses. There are 120 CSS-to-core barrel bolts, which represents a significant level of redundancy, i.e., a large number of bolts would have to fail before the internals would drop. TMI-1 original bolts are fabricated with X-750. The asterisks (*) indicate that the consequences are as a result of a cascading (or dependent) failure. As some bolts fail, the remaining bolts will be subject to a greater load, which will increase their likelihood of failure. Reactor vessel guide lugs are designed to halt the fall of the internals/core (onto the shock pads) and not affect control rod insertion. However, note, that while not in the PWR internals scope, the guide lugs need to maintain functionality to stop the fall of the internals.
					If a significant number of bolts were degraded, the internals (core barrel assembly, core, and lower internals assembly) would drop (and increase bypass flow).	D	3*	4*	Yes. A number of detection means are possible: neutron noises, radioactivity alarms, loose parts monitoring.	
		Fatigue (Thermal Stress Relaxation/ Creep)	Cracking/ Fracture	Loss of mechanical joint integrity	No effect if a small number of bolts are degraded. If locking clips fail (S.5 has no credible degradation mechanism), broken pieces from locking clips and/or bolts could result in loose parts. Also applies to wear.	B	1	2	No.	There are 120 CSS-to-core barrel bolts, which represents a significant level of redundancy, i.e., a large number of bolts would have to fail before the internals would drop. The asterisks (*) indicate that the consequences are as a result of a cascading (or dependent) failure. As some bolts fail, the remaining bolts will be subject to a greater load, which will increase their likelihood of failure. (Only applies to ONS, since ANO-1 has replaced all but six bolts, and Davis-Besse all but three bolts.) Reactor vessel guide lugs are designed to halt the fall of the internals/core (onto the shock pads) and not affect control rod insertion. However, note, that while not in the PWR internals scope, the guide lugs need to maintain functionality to stop the fall of the internals.
					If a significant number of bolts were degraded, the internals (core barrel assembly, core, and lower internals assembly) would drop (and increase bypass flow).	B	3*	4*	Yes. A number of detection means are possible: neutron noises, radioactivity alarms, loose parts monitoring.	
		Wear (Thermal Stress Relaxation/Creep)	Loss of material	Loss of mechanical joint integrity	No effect if a small number of bolts are degraded.	B	1	2	No.	There are 120 CSS-to-core barrel bolts, which represents a significant level of redundancy, i.e., a large number of bolts would have to fail before the internals would drop. The asterisks (*) indicate that the consequences are as a result of a cascading (or dependent) failure. As some bolts fail, the remaining bolts will be subject to a greater load, which will increase their likelihood of failure. (Only applies to ONS, since ANO-1 has replaced all but six bolts, and Davis-Besse all but three bolts.) Reactor vessel guide lugs are designed to halt the fall of the internals/core (onto the shock pads) and not affect control rod insertion. However, note, that while not in the PWR internals scope, the guide lugs need to maintain functionality to stop the fall of the internals.
					If a significant number of bolts were degraded, the internals (core barrel assembly, core, and lower internals assembly) would drop (and increase bypass flow).	B	3*	4*	Yes. A number of detection means are possible: neutron noises, radioactivity alarms, loose parts monitoring.	

Failure Modes, Effects, and Criticality Analysis (FMECA) Table

Component Item Name	FMECA Identifier	Degradation Mechanism	Failure Mode	Failure Effect (Local)	Failure Effect (Global)	Susceptibility	Severity of Consequences (Safety)	Severity of Consequences (Economic)	Detectable	Comments
Replacement CSS-to-Core Barrel Bolts (ANO-1, CR-3)	S.4c	SCC	Cracking/ Fracture	Localized degradation of structural integrity	No effect if a small number of bolts are degraded.	C	1	2	No.	These bolts subject to SCC due to the material type and stresses. There are 120 CSS-to-core barrel bolts, which represents a significant level of redundancy, i.e., a large number of bolts would have to fail before the internals would drop. Replacement bolts have been designed to reduce stress concentration and are subjected to reduced load (reducing susceptibility to SCC). The asterisks (*) indicate that the consequences are as a result of a cascading (or dependent) failure. As some bolts fail, the remaining bolts will be subject to a greater load, which will increase their likelihood of failure. Reactor vessel guide lugs are designed to halt the fall of the internals/core (onto the shock pads) and not affect control rod insertion. However, note, that while not in the PWR internals scope, the guide lugs need to maintain functionality to stop the fall of the internals.
					If a significant number of bolts were degraded, the internals (core barrel assembly, core, and lower internals assembly) would drop (and increase bypass flow).	C	3*	4*	Yes. A number of detection means are possible: neutron noises, radioactivity alarms, loose parts monitoring.	
		Fatigue (Thermal Stress Relaxation/ Creep)	Cracking/ Fracture	Loss of mechanical joint integrity	No effect if a small number of bolts are degraded. If locking clips fail (S.5 has no credible degradation mechanism), broken pieces from locking clips and/or bolts could result in loose parts. Also applies to wear.	B	1	2	No.	There are 120 CSS-to-core barrel bolts, which represents a significant level of redundancy, i.e., a large number of bolts would have to fail before the internals would drop. The asterisks (*) indicate that the consequences are as a result of a cascading (or dependent) failure. As some bolts fail, the remaining bolts will be subject to a greater load, which will increase their likelihood of failure. (Only applies to ONS, since ANO-1 has replaced all but six bolts, and Davis-Besse all but three bolts.) Reactor vessel guide lugs are designed to halt the fall of the internals/core (onto the shock pads) and not affect control rod insertion. However, note, that while not in the PWR internals scope, the guide lugs need to maintain functionality to stop the fall of the internals.
					If a significant number of bolts were degraded, the internals (core barrel assembly, core, and lower internals assembly) would drop (and increase bypass flow).	B	3*	4*	Yes. A number of detection means are possible: neutron noises, radioactivity alarms, loose parts monitoring.	
		Wear (Thermal Stress Relaxation/Creep)	Loss of material	Loss of mechanical joint integrity	No effect if a small number of bolts are degraded.	B	1	2	No.	There are 120 CSS-to-core barrel bolts, which represents a significant level of redundancy, i.e., a large number of bolts would

Component Item Name	FMECA Identifier	Degradation Mechanism	Failure Mode	Failure Effect (Local)	Failure Effect (Global)	Susceptibility	Severity of Consequences (Safety)	Severity of Consequences (Economic)	Detectable	Comments
					If a significant number of bolts were degraded, the internals (core barrel assembly, core, and lower internals assembly) would drop (and increase bypass flow).	B	3*	4*	Yes. A number of detection means are possible: neutron noises, radioactivity alarms, loose parts monitoring..	have to fail before the internals would drop. The asterisks (*) indicate that the consequences are as a result of a cascading (or dependent) failure. As some bolts fail, the remaining bolts will be subject to a greater load, which will increase their likelihood of failure. (Only applies to ONS, since ANO-1 has replaced all but six bolts, and Davis-Besse all but three bolts.) Reactor vessel guide lugs are designed to halt the fall of the internals/core (onto the shock pads) and not affect control rod insertion. However, note, that while not in the PWR internals scope, the guide lugs need to maintain functionality to stop the fall of the internals.
Replacement CSS-to-Core Barrel Bolts (DB)	S.4d	SCC	Cracking/ Fracture	Localized degradation of structural integrity	No effect if a small number of bolts are degraded.	C	1	2	No.	These bolts subject to SCC due to the material type and stresses. There are 120 CSS-to-core barrel bolts, which represents a significant level of redundancy, i.e., a large number of bolts would have to fail before the internals would drop. Replacement bolts have been designed to reduce stress concentration and are subjected to reduced load (reducing susceptibility to SCC). The asterisks (*) indicate that the consequences are as a result of a cascading (or dependent) failure. As some bolts fail, the remaining bolts will be subject to a greater load, which will increase their likelihood of failure. Reactor vessel guide lugs are designed to halt the fall of the internals/core (onto the shock pads) and not affect control rod insertion. However, note, that while not in the PWR internals scope, the guide lugs need to maintain functionality to stop the fall of the internals.
					If a significant number of bolts were degraded, the internals (core barrel assembly, core, and lower internals assembly) would drop (and increase bypass flow).	C	3*	4*	Yes. A number of detection means are possible: neutron noises, radioactivity alarms, loose parts monitoring.	
		Fatigue (Thermal Stress Relaxation/ Creep)	Cracking/ Fracture	Loss of mechanical joint integrity	No effect if a small number of bolts are degraded. If locking clips fail (S.5 has no credible degradation mechanism), broken pieces from locking clips and/or bolts could result in loose parts. Also applies to wear.	B	1	2	No.	There are 120 CSS-to-core barrel bolts, which represents a significant level of redundancy, i.e., a large number of bolts would have to fail before the internals would drop. The asterisks (*) indicate that the consequences are as a result of a cascading (or dependent) failure. As some bolts fail, the remaining bolts will be subject to a greater load, which will increase their likelihood of failure. (Only applies to ONS, since ANO-1 has replaced all but six bolts, and Davis-Besse all but three bolts.) Reactor vessel guide lugs are designed to halt the fall of the internals/core (onto the shock pads) and not affect control rod insertion. However, note, that while not in the PWR internals scope, the guide lugs need to maintain functionality to stop the fall of the internals.
					If a significant number of bolts were degraded, the internals (core barrel assembly, core, and lower internals assembly) would drop (and increase bypass flow).	B	3*	4*	Yes. A number of detection means are possible: neutron noises, radioactivity alarms, loose parts monitoring.	

Failure Modes, Effects, and Criticality Analysis (FMECA) Table

Component Item Name	FMECA Identifier	Degradation Mechanism	Failure Mode	Failure Effect (Local)	Failure Effect (Global)	Susceptibility	Severity of Consequences (Safety)	Severity of Consequences (Economic)	Detectable	Comments
		Wear (Thermal Stress Relaxation/Creep)	Loss of material	Loss of mechanical joint integrity	No effect if a small number of bolts are degraded.	B	1	2	No.	There are 120 CSS-to-core barrel bolts, which represents a significant level of redundancy, i.e., a large number of bolts would have to fail before the internals would drop.
					If a significant number of bolts were degraded, the internals (core barrel assembly, core, and lower internals assembly) would drop (and increase bypass flow).	B	3*	4*	Yes. A number of detection means are possible: neutron noises, radioactivity alarms, loose parts monitoring.	The asterisks (*) indicate that the consequences are as a result of a cascading (or dependent) failure. As some bolts fail, the remaining bolts will be subject to a greater load, which will increase their likelihood of failure. (Only applies to ONS, since ANO-1 has replaced all but six bolts, and Davis-Besse all but three bolts.) Reactor vessel guide lugs are designed to halt the fall of the internals/core (onto the shock pads) and not affect control rod insertion. However, note, that while not in the PWR internals scope, the guide lugs need to maintain functionality to stop the fall of the internals.
Original CSS-to-Core Barrel Bolts Locking Clips (ANO-1, DB, ONS, TMI-1)	S.5a	No credible degradation mechanism	n/a	n/a	n/a	A	Not evaluated	Not evaluated		The screening analysis shows that none of the screening parameter values are exceeded (Category A) and therefore the likelihood of any consequential age-related degradation is extremely small. Accordingly, this component item is only evaluated (and screened) on a susceptibility basis. (No evaluation of consequence is performed.)
Replacement CSS-to-Core Barrel Bolts Locking Cups (ANO-1, CR-3, DB)	S.5b	No credible degradation mechanism	n/a	n/a	n/a	A	Not evaluated	Not evaluated		The screening analysis shows that none of the screening parameter values are exceeded (Category A) and therefore the likelihood of any consequential age-related degradation is extremely small. Accordingly, this component item is only evaluated (and screened) on a susceptibility basis. (No evaluation of consequence is performed.)
Replacement CSS-to-Core Barrel Bolts Tie Plates (ANO-1, CR-3, DB)	S.5c	No credible degradation mechanism	n/a	n/a	n/a	A	Not evaluated	Not evaluated		The screening analysis shows that none of the screening parameter values are exceeded (Category A) and therefore the likelihood of any consequential age-related degradation is extremely small. Accordingly, this component item is only evaluated (and screened) on a susceptibility basis. (No evaluation of consequence is performed.)
Outlet Nozzles (ANO-1, ONS-1, ONS-2, TMI-1)	S.6a	SCC	Cracking/ Fracture	Localized degradation of structural integrity (at the HAZ)	No operational effects.	B	1	2	No.	Economic consequences driven by analysis and/or repairs.
Outlet Nozzles (ONS-3, DB)	S.6b	Thermal Aging Embrittlement	Less flaw tolerant	More susceptible to crack growth when subjected to an external load with a flaw	More severe cracking.	C	1	2	No.	More susceptible to crack growth from an external load/force. Decrease in toughness.
Vent Valve Nozzles	S.7	SCC	Cracking/ Fracture	Localized degradation of structural integrity (at the HAZ)	No operational effects.	B	1	2	No.	Economic consequences driven by analysis and/or repairs.
Vent Valve Guide Blocks	S.8	SCC	Cracking/ Fracture	Localized degradation of structural integrity (at the HAZ)	No operational effects.	B	1	2	No.	Guide blocks are used to provide vent valve alignment. Economic consequences driven by analysis and/or repairs.

Component Item Name	FMECA Identifier	Degradation Mechanism	Failure Mode	Failure Effect (Local)	Failure Effect (Global)	Susceptibility	Severity of Consequences (Safety)	Severity of Consequences (Economic)	Detectable	Comments
Vent Valve Body	S.9	No credible degradation mechanism	n/a	n/a	n/a	A	Not evaluated	Not evaluated		The screening analysis shows that none of the screening parameter values are exceeded (Category A) and therefore the likelihood of any consequential age-related degradation is extremely small. Accordingly, this component item is only evaluated (and screened) on a susceptibility basis. (No evaluation of consequence is performed.)
Vent Valve Retaining Rings	S.10	Thermal Aging Embrittlement	Less flaw tolerant	More susceptible to crack growth when subjected to an external load with a flaw	None	C	1	2	No.	More susceptible to crack growth from an external load/force. Decrease in toughness. Economic consequences driven by analysis and/or repairs.
Vent Valve Jack Screws	S.11	Fatigue (Thermal Stress Relaxation/ Creep)	Cracking/ Fracture	Loss of mechanical joint integrity	No operational effects.	B	1	2	No.	
		Wear (Thermal Stress Relaxation/Creep)	Loss of material	Loss of mechanical joint integrity	No operational effects.	B	1	2	No.	
Disc	S.12	Thermal Aging Embrittlement	Less flaw tolerant	More susceptible to crack growth when subjected to an external load with a flaw	None	C	1	2	No.	More susceptible to crack growth from an external load/force. Decrease in toughness. Economic consequences driven by analysis and/or repairs.
Disc Bushing	S.13	No credible degradation mechanism	n/a	n/a	n/a	A	Not evaluated	Not evaluated		The screening analysis shows that none of the screening parameter values are exceeded (Category A) and therefore the likelihood of any consequential age-related degradation is extremely small. Accordingly, this component item is only evaluated (and screened) on a susceptibility basis. (No evaluation of consequence is performed.)
Disc Shaft/Hinge Pin	S.14	Thermal Aging Embrittlement	Less flaw tolerant	More susceptible to crack growth when subjected to an external load with a flaw	None	C	1	2	No.	More susceptible to crack growth from an external load/force. Decrease in toughness. Corrosion locking the hinge is not an issue, since the hinge is exercised every refueling outage. Economic consequences driven by analysis and/or repairs.
Round Bars	S.15	SCC	Cracking/ Fracture	Localized degradation of structural integrity (at the HAZ)	Loose parts. Loss of integrity of most or all of the round bars in both the Plenum Cylinder Assembly and the Core Support Shield Assembly could restrict coolant flow during a response to a LOCA. See item P.2.5.	B/I B/I	Not evaluated Not evaluated	Not evaluated Not evaluated	Yes. Loose Parts Monitoring System (LPMS)	In the event of a hot leg LOCA, the round bars will prevent the plenum assembly and the core support shield assembly from collapsing.
Flow Deflectors	S.16	SCC	Cracking/ Fracture	Localized degradation of structural integrity (at the HAZ)	No operation effects.	B	1	1	No.	The sides of the flow detectors have been machined down.

Failure Modes, Effects, and Criticality Analysis (FMECA) Table

Component Item Name	FMECA Identifier	Degradation Mechanism	Failure Mode	Failure Effect (Local)	Failure Effect (Global)	Susceptibility	Severity of Consequences (Safety)	Severity of Consequences (Economic)	Detectable	Comments
Lifting Lugs	S.17	SCC	Cracking/ Fracture	Localized degradation of structural integrity (at the HAZ)	No operational effects; inability to lift internals during refueling	B	1	3	Yes.	Inability to remove internals could significantly delay a refueling outage while repairs were being performed. If weld(s) fail while internals is being lifted, the resulting fall could damage other internals component items, and/or damage fuel (radiological consequences).

Table A-3
Core Barrel Assembly (B)

Component Item Name	FMECA Identifier	Degradation Mechanism	Failure Mode	Failure Effect (Local)	Failure Effect (Global)	Susceptibility	Severity of Consequences (Safety)	Severity of Consequences (Economic)	Detectable	Comments
Core Barrel Assembly (B)	B	Function	Direct reactor coolant flow, support the lower internals assembly Reduce the amount of radiation that reaches the reactor vessel (thermal shield)							
Core Barrel Cylinder	B.1	SCC	Cracking/ Fracture	Localized degradation of structural integrity	No operational effects.	B	1	2		SCC cracking is tight. Does not permit any significant flow diversion. SCC cracking potential is at the heat-affected zone (HAZ) and the flange welds (see items S.2 and S.3).
		Fatigue	Cracking/ Fracture	Localized degradation of structural integrity; leakage	No operational effects. There is a small delta-p and other holes in the core barrel cylinder.	A	Not evaluated	Not evaluated		Low susceptibility is supported by no known operating experience of fatigue, and design criteria contain a significant amount of margin.
		Irradiation Embrittlement	Less flaw tolerant	More susceptible to crack growth when subjected to an external load with a flaw	No operational effects.	C	1	2		More susceptible to crack growth from an external load/force. Decrease in toughness.
Top Flange	B.2	SCC	Cracking/ Fracture	Localized degradation of structural integrity (at the HAZ)	No operational effects. Critical flaw size (crack) around the weld separating the top flange from the core barrel cylinder (causing the internals and core to fall and increase bypass flow).	B **B/I**	1 **Not evaluated**	2 **Not evaluated**	No.	Reactor vessel guide lugs are designed to halt the fall of the internals/core (onto the shock pads) and not affect control rod insertion. However, note, that while not in the PWR internals scope, the guide lugs need to maintain functionality to stop the fall of the internals.
		Irradiation Embrittlement	Less flaw tolerant	More susceptible to crack growth when subjected to an external load with a flaw	No operational effects. Top flange is outside the belt region; fluence levels are not high enough for severe embrittlement.	B	1	2		More susceptible to crack growth from an external load/force. Decrease in toughness.
Bottom Flange	B.3	SCC	Cracking/ Fracture	Localized degradation of structural integrity (at the HAZ)	No operational effects. Critical flaw size (crack) around the weld separating the core barrel cylinder from the bottom flange (causing the internals and core to fall and increase bypass flow).	B **B/I**	1 **Not evaluated**	2 **Not evaluated**	No.	Reactor vessel guide lugs are designed to halt the fall of the internals/core (onto the shock pads) and not affect control rod insertion. However, note, that while not in the PWR internals scope, the guide lugs need to maintain functionality to stop the fall of the internals.
		Irradiation Embrittlement	Less flaw tolerant	More susceptible to crack growth when subjected to an external load with a flaw	No operational effects. Bottom flange is outside the belt region; fluence levels are not high enough for severe embrittlement.	B	1	2		More susceptible to crack growth from an external load/force. Decrease in toughness.
Original Lower Grid Assembly-to-Core	B.4a	SCC	Cracking/ Fracture	Localized degradation of structural integrity	No effect if a small number of bolts are degraded.	D	1	2	No.	These bolts subject to SCC due to the material type and stresses. There are 108 CSS-to-core barrel bolts, which represents a

Failure Modes, Effects, and Criticality Analysis (FMECA) Table

Component Item Name	FMECA Identifier	Degradation Mechanism	Failure Mode	Failure Effect (Local)	Failure Effect (Global)	Susceptibility	Severity of Consequences (Safety)	Severity of Consequences (Economic)	Detectable	Comments
Barrel Bolts (ANO-1, CR-3, DB, ONS)					If a significant number of bolts were degraded, the internals (lower internals assembly) would drop (and increase bypass flow).	D	3*	4*	Yes. A number of detection means are possible: neutron noises, radioactivity alarms, loose parts monitoring. Fuel assemblies will tend to move up and down.	significant level of redundancy, i.e., a large number of bolts would have to fail before the internals would drop. The asterisks (*) indicate that the consequences are as a result of a cascading (or dependent) failure. As some bolts fail, the remaining bolts will be subject to a greater load, which will increase their likelihood of failure. (Does not apply Davis-Besse or CR-3 since each has replaced 60 bolts.) Reactor vessel guide lugs are designed to halt the fall of the internals/core (onto the shock pads) and not affect control rod insertion. However, note, that while not in the PWR internals scope, the guide lugs need to maintain functionality to stop the fall of the internals.
		Fatigue (Thermal Stress Relaxation/ Creep)	Cracking/ Fracture	Loss of mechanical joint integrity	No effect if a small number of bolts are degraded. If locking clips fail (B.5 has no credible degradation mechanism), broken pieces from locking clips and/or bolts could result in loose parts. Also applies to wear.	B	1	2	No.	There are 108 CSS-to-core barrel bolts, which represents a significant level of redundancy, i.e., a large number of bolts would have to fail before the internals would drop. The asterisks (*) indicate that the consequences are as a result of a cascading (or dependent) failure. As some bolts fail, the remaining bolts will be subject to a greater load, which will increase their likelihood of failure. (Does not apply Davis-Besse or CR-3 since each has replaced 60 bolts.) Reactor vessel guide lugs are designed to halt the fall of the internals/core (onto the shock pads) and not affect control rod insertion. However, note, that while not in the PWR internals scope, the guide lugs need to maintain functionality to stop the fall of the internals.
					If a significant number of bolts were degraded, the internals (lower internals assembly) would drop (and increase bypass flow).	B	3*	4*	Yes. A number of detection means are possible: neutron noises, radioactivity alarms, loose parts monitoring. Fuel assemblies will tend to move up and down.	
		Wear (Thermal Stress Relaxation/Creep)	Loss of material	Loss of mechanical joint integrity	No effect if a small number of bolts are degraded.	B	1	2	No.	There are 108 CSS-to-core barrel bolts, which represents a significant level of redundancy, i.e., a large number of bolts would have to fail before the internals would drop. The asterisks (*) indicate that the consequences are as a result of a cascading (or dependent) failure. As some bolts fail, the remaining bolts will be subject to a greater load, which will increase their likelihood of failure. (Does not apply Davis-Besse or CR-3 since each has replaced 60 bolts.) Reactor vessel guide lugs are designed to halt the fall of the internals/core (onto the shock pads) and not affect control rod insertion. However, note, that while not in the PWR internals scope, the guide lugs need to maintain functionality to stop the fall of the internals.
					If a significant number of bolts were degraded, the internals (core barrel assembly, core, and lower internals assembly) would drop (and increase bypass flow).	B	3*	4*	Yes. A number of detection means are possible: neutron noises, radioactivity alarms, loose parts monitoring. Fuel assemblies will tend to move up and down.	

Component Item Name	FMECA Identifier	Degradation Mechanism	Failure Mode	Failure Effect (Local)	Failure Effect (Global)	Susceptibility	Severity of Consequences (Safety)	Severity of Consequences (Economic)	Detectable	Comments
Original Lower Grid Assembly-to-Core Barrel Bolts (ONS-1)	B.4b	Fatigue (Operational Cyclic Loading and Thermal Stress Relaxation/Creep)	Cracking/ Fracture	Loss of mechanical joint integrity	No effect if a small number of bolts are degraded. If locking clips fail (B.5 has no credible degradation mechanism), broken pieces from locking clips and/or bolts could result in loose parts. Also applies to wear.	B	1	2	No.	There are 108 CSS-to-core barrel bolts, which represents a significant level of redundancy, i.e., a large number of bolts would have to fail before the internals would drop. The asterisks (*) indicate that the consequences are as a result of a cascading (or dependent) failure. As some bolts fail, the remaining bolts will be subject to a greater load, which will increase their likelihood of failure. Reactor vessel guide lugs are designed to halt the fall of the internals/core (onto the shock pads) and not affect control rod insertion. However, note, that while not in the PWR internals scope, the guide lugs need to maintain functionality to stop the fall of the internals.
					If a significant number of bolts were degraded, the internals (lower internals assembly) would drop (and increase bypass flow).	B	3*	4*	Yes. A number of detection means are possible: neutron noises, radioactivity alarms, loose parts monitoring. Fuel assemblies will tend to move up and down.	
		Wear (Thermal Stress Relaxation/Creep)	Loss of material	Loss of mechanical joint integrity	No effect if a small number of bolts are degraded.	B	1	2	No.	There are 108 CSS-to-core barrel bolts, which represents a significant level of redundancy, i.e., a large number of bolts would have to fail before the internals would drop. The asterisks (*) indicate that the consequences are as a result of a cascading (or dependent) failure. As some bolts fail, the remaining bolts will be subject to a greater load, which will increase their likelihood of failure. Reactor vessel guide lugs are designed to halt the fall of the internals/core (onto the shock pads) and not affect control rod insertion. However, note, that while not in the PWR internals scope, the guide lugs need to maintain functionality to stop the fall of the internals.
					If a significant number of bolts were degraded, the internals (core barrel assembly, core, and lower internals assembly) would drop (and increase bypass flow).	B	3*	4*	Yes. A number of detection means are possible: neutron noises, radioactivity alarms, loose parts monitoring.	
Original Lower Grid Assembly-to-Core Barrel Bolts (TMI-1)	B.4c	SCC	Cracking/ Fracture	Localized degradation of structural integrity	No effect if a small number of bolts are degraded.	D	1	2	No.	These bolts subject to SCC due to the material type and stresses. There are 108 CSS-to-core barrel bolts, which represents a significant level of redundancy, i.e., a large number of bolts would have to fail before the internals would drop. The asterisks (*) indicate that the consequences are as a result of a cascading (or dependent) failure. As some bolts fail, the remaining bolts will be subject to a greater load, which will increase their likelihood of failure. Reactor vessel guide lugs are designed to halt the fall of the internals/core (onto the shock pads) and not affect control rod insertion. However, note, that while not in the PWR internals scope, the guide lugs need to maintain functionality to stop the fall of the internals.
					If a significant number of bolts were degraded, the internals (lower internals assembly) would drop (and increase bypass flow).	D	3*	4*	Yes. A number of detection means are possible: neutron noises, radioactivity alarms, loose parts monitoring. Fuel assemblies will tend to move up and down.	

Failure Modes, Effects, and Criticality Analysis (FMECA) Table

Component Item Name	FMECA Identifier	Degradation Mechanism	Failure Mode	Failure Effect (Local)	Failure Effect (Global)	Susceptibility	Severity of Consequences (Safety)	Severity of Consequences (Economic)	Detectable	Comments
		Fatigue (Operational Cyclic Loading and Thermal Stress Relaxation/Creep)	Cracking/ Fracture	Loss of mechanical joint integrity	No effect if a small number of bolts are degraded. If locking clips fail (B.5 has no credible degradation mechanism), broken pieces from locking clips and/or bolts could result in loose parts. Also applies to wear.	B	1	2	No.	There are 108 CSS-to-core barrel bolts, which represents a significant level of redundancy, i.e., a large number of bolts would have to fail before the internals would drop. The asterisks (*) indicate that the consequences are as a result of a cascading (or dependent) failure. As some bolts fail, the remaining bolts will be subject to a greater load, which will increase their likelihood of failure. Reactor vessel guide lugs are designed to halt the fall of the internals/core (onto the shock pads) and not affect control rod insertion. However, note, that while not in the PWR internals scope, the guide lugs need to maintain functionality to stop the fall of the internals.
					If a significant number of bolts were degraded, the internals (lower internals assembly) would drop (and increase bypass flow).	B	3*	4*	Yes. A number of detection means are possible: neutron noises, radioactivity alarms, loose parts monitoring. Fuel assemblies will tend to move up and down.	
		Wear (Thermal Stress Relaxation/Creep)	Loss of material	Loss of mechanical joint integrity	No effect if a small number of bolts are degraded.	B	1	2	No.	There are 108 CSS-to-core barrel bolts, which represents a significant level of redundancy, i.e., a large number of bolts would have to fail before the internals would drop. The asterisks (*) indicate that the consequences are as a result of a cascading (or dependent) failure. As some bolts fail, the remaining bolts will be subject to a greater load, which will increase their likelihood of failure. Reactor vessel guide lugs are designed to halt the fall of the internals/core (onto the shock pads) and not affect control rod insertion. However, note, that while not in the PWR internals scope, the guide lugs need to maintain functionality to stop the fall of the internals.
					If a significant number of bolts were degraded, the internals (core barrel assembly, core, and lower internals assembly) would drop (and increase bypass flow).	B	3*	4*	Yes. A number of detection means are possible: neutron noises, radioactivity alarms, loose parts monitoring.	
Replacement Lower Grid Assembly-to-Core Barrel Bolts (CR-3, DB)	B.4d	SCC	Cracking/ Fracture	Localized degradation of structural integrity	No effect if a small number of bolts are degraded.	C	1	2	No.	These bolts subject to SCC due to the material type and stresses. There are 108 CSS-to-core barrel bolts, which represents a significant level of redundancy, i.e., a large number of bolts would have to fail before the internals would drop. Replacement bolts have been subjected to heat treatment and reduced load (slightly reducing susceptibility to SCC). The asterisks (*) indicate that the consequences are as a result of a cascading (or dependent) failure. As some bolts fail, the remaining bolts will be subject to a greater load, which will increase their likelihood of failure. Reactor vessel guide lugs are designed to halt the fall of the internals/core (onto the shock pads) and not affect control rod insertion. However, note, that while not in the PWR internals scope, the guide lugs need to maintain functionality to stop the fall of the internals.
					If a significant number of bolts were degraded, the internals (lower internals assembly) would drop (and increase bypass flow).	C	3*	4*	Yes. A number of detection means are possible: neutron noises, radioactivity alarms, loose parts monitoring. Fuel assemblies will tend to move up and down.	

Component Item Name	FMECA Identifier	Degradation Mechanism	Failure Mode	Failure Effect (Local)	Failure Effect (Global)	Susceptibility	Severity of Consequences (Safety)	Severity of Consequences (Economic)	Detectable	Comments
		Fatigue (Thermal Stress Relaxation/ Creep)	Cracking/ Fracture	Loss of mechanical joint integrity	No effect if a small number of bolts are degraded. If locking clips fail (B.5 has no credible degradation mechanism), broken pieces from locking clips and/or bolts could result in loose parts. Also applies to wear.	B	1	2	No.	There are 108 CSS-to-core barrel bolts, which represents a significant level of redundancy, i.e., a large number of bolts would have to fail before the internals would drop. The asterisks (*) indicate that the consequences are as a result of a cascading (or dependent) failure. As some bolts fail, the remaining bolts will be subject to a greater load, which will increase their likelihood of failure. (Does not apply Davis-Besse or CR-3 since each has replaced 60 bolts.) Reactor vessel guide lugs are designed to halt the fall of the internals/core (onto the shock pads) and not affect control rod insertion. However, note, that while not in the PWR internals scope, the guide lugs need to maintain functionality to stop the fall of the internals.
					If a significant number of bolts were degraded, the internals (lower internals assembly) would drop (and increase bypass flow).	B	3*	4*	Yes. A number of detection means are possible: neutron noises, radioactivity alarms, loose parts monitoring. Fuel assemblies will tend to move up and down.	
		Wear (Thermal Stress Relaxation/Creep)	Loss of material	Loss of mechanical joint integrity	No effect if a small number of bolts are degraded.	B	1	2	No.	There are 108 CSS-to-core barrel bolts, which represents a significant level of redundancy, i.e., a large number of bolts would have to fail before the internals would drop. The asterisks (*) indicate that the consequences are as a result of a cascading (or dependent) failure. As some bolts fail, the remaining bolts will be subject to a greater load, which will increase their likelihood of failure. (Does not apply Davis-Besse or CR-3 since each has replaced 60 bolts.) Reactor vessel guide lugs are designed to halt the fall of the internals/core (onto the shock pads) and not affect control rod insertion. However, note, that while not in the PWR internals scope, the guide lugs need to maintain functionality to stop the fall of the internals.
					If a significant number of bolts were degraded, the internals (core barrel assembly, core, and lower internals assembly) would drop (and increase bypass flow).	B	3*	4*	Yes. A number of detection means are possible: neutron noises, radioactivity alarms, loose parts monitoring. Fuel assemblies will tend to move up and down.	
Original Lower Grid Assembly-to-Core Barrel Bolts Locking Clips (ANO-1, CR-3, DB, ONS, TMI-1)	B.5a	No credible degradation mechanism	n/a	n/a	n/a	A	Not evaluated	Not evaluated		The screening analysis shows that none of the screening parameter values are exceeded (Category A) and therefore the likelihood of any consequential age-related degradation is extremely small. Accordingly, this component item is only evaluated (and screened) on a susceptibility basis. (No evaluation of consequence is performed.)
Replacement Lower Grid Assembly-to-Core Barrel Bolts Locking Cups & Tie Plates (CR-3, DB)	B.5b	No credible degradation mechanism	n/a	n/a	n/a	A	Not evaluated	Not evaluated		The screening analysis shows that none of the screening parameter values are exceeded (Category A) and therefore the likelihood of any consequential age-related degradation is extremely small. Accordingly, this component item is only evaluated (and screened) on a susceptibility basis. (No evaluation of consequence is performed.)

Component Item Name	FMECA Identifier	Degradation Mechanism	Failure Mode	Failure Effect (Local)	Failure Effect (Global)	Susceptibility	Severity of Consequences (Safety)	Severity of Consequences (Economic)	Detectable	Comments
Thermal Shield-to-Core Barrel Bolts (ANO-1,CR-3, DB, ONS)	B.6a	SCC	Cracking/ Fracture	Localized degradation of structural integrity	No operational effects. Restraint blocks are secured with three welded dowels and three bolts (with locking clips). If enough bolts fail, thermal shield may "wiggle."	C	1	2	No.	These bolts are also known as the upper thermal shield bolts, and are used to fasten the shim, the inner "B," and the outer "A" blocks to the core barrel cylinder. The "A" and "B" blocks form the notch that holds the thermal shield and allows for thermal expansion. Each restraint block is positioned and secured with three dowels (captured by weld plugs) and three bolts (with locking clips).
		Fatigue (Thermal Stress Relaxation/ Creep)	Cracking/ Fracture	Loss of mechanical joint integrity	No operational effects. Restraint blocks are secured with three welded dowels and three bolts (with locking clips). If enough bolts fail, thermal shield may "wiggle." If locking clips fail (B.7 has no credible degradation mechanism identified), broken pieces from locking clips and/or bolts could result in loose parts. Also applies to wear.	B	1	2	No.	These bolts are also known as the upper thermal shield bolts, and are used to fasten the shim, the inner "B," and the outer "A" blocks to the core barrel cylinder. The "A" and "B" blocks form the notch that holds the thermal shield and allows for thermal expansion. Each restraint block is positioned and secured with three dowels (captured by weld plugs) and three bolts (with locking clips).
		Wear (Thermal Stress Relaxation/Creep)	Loss of material	Loss of mechanical joint integrity	No operational effects. Restraint blocks are secured with three welded dowels and three bolts (with locking clips). If enough bolts fail, thermal shield may "wiggle."	B	1	2	No.	These bolts are also known as the upper thermal shield bolts, and are used to fasten the shim, the inner "B," and the outer "A" blocks to the core barrel cylinder. The "A" and "B" blocks form the notch that holds the thermal shield and allows for thermal expansion. Each restraint block is positioned and secured with three dowels (captured by weld plugs) and three bolts (with locking clips).
Thermal Shield-to-Core Barrel Bolts (TMI-1)	B.6b	SCC	Cracking/ Fracture	Localized degradation of structural integrity	No operational effects. Restraint blocks are secured with three welded dowels and three bolts (with locking clips). If enough bolts fail, thermal shield may "wiggle."	C	1	2	No.	These bolts are also known as the upper thermal shield bolts, and are used to fasten the shim, the inner "B," and the outer "A" blocks to the core barrel cylinder. The "A" and "B" blocks form the notch that holds the thermal shield and allows for thermal expansion. Each restraint block is positioned and secured with three dowels (captured by weld plugs) and three bolts (with locking clips).
		Fatigue (Thermal Stress Relaxation/ Creep)	Cracking/ Fracture	Loss of mechanical joint integrity	No operational effects. Restraint blocks are secured with three welded dowels and three bolts (with locking clips). If enough bolts fail, thermal shield may "wiggle." If locking clips fail (B.7 has no credible degradation mechanism identified), broken pieces from locking clips and/or bolts could result in loose parts. Also applies to wear.	B	1	2	No.	These bolts are also known as the upper thermal shield bolts, and are used to fasten the shim, the inner "B," and the outer "A" blocks to the core barrel cylinder. The "A" and "B" blocks form the notch that holds the thermal shield and allows for thermal expansion. Each restraint block is positioned and secured with three dowels (captured by weld plugs) and three bolts (with locking clips).

Component Item Name	FMECA Identifier	Degradation Mechanism	Failure Mode	Failure Effect (Local)	Failure Effect (Global)	Susceptibility	Severity of Consequences (Safety)	Severity of Consequences (Economic)	Detectable	Comments
		Wear (Thermal Stress Relaxation/Creep)	Loss of material	Loss of mechanical joint integrity	No operational effects. Restraint blocks are secured with three welded dowels and three bolts (with locking clips). If enough bolts fail, thermal shield may "wiggle."	B	1	2	No.	These bolts are also known as the upper thermal shield bolts, and are used to fasten the shim, the inner "B," and the outer "A" blocks to the core barrel cylinder. The "A" and "B" blocks form the notch that holds the thermal shield and allows for thermal expansion. Each restraint block is positioned and secured with three dowels (captured by weld plugs) and three bolts (with locking clips).
Thermal Shield-to-Core Barrel Bolts Locking Clips (ANO-1,CR-3, DB, ONS, TMI-1)	B.7	No credible degradation mechanism	n/a	n/a	n/a	A	Not evaluated	Not evaluated		The screening analysis shows that none of the screening parameter values are exceeded (Category A) and therefore the likelihood of any consequential age-related degradation is extremely small. Accordingly, this component item is only evaluated (and screened) on a susceptibility basis. (No evaluation of consequence is performed.)
Replacement Surveillance Specimen Holder Tube-to-Thermal Shield Studs/Nuts (CR-3)	B.8a	SCC	Cracking/ Fracture	Localized degradation of structural integrity	No operational effects. Possibly minor damage to the thermal shield.	C	1	2	No.	
		Fatigue (Thermal Stress Relaxation/ Creep)	Cracking/ Fracture	Loss of mechanical joint integrity	No operational effects. Possibly minor damage to the thermal shield. If locking devices fail (B.9 has no credible degradation mechanism identified), broken pieces from locking devices and/or bolts could result in loose parts. Also applies to wear.	B	1	2	No.	
		Wear (Thermal Stress Relaxation/Creep)	Loss of material	Loss of mechanical joint integrity	No operational effects. Possibly minor damage to the thermal shield.	B	1	2	No.	
Replacement Surveillance Specimen Holder Tube-to-Thermal Shield Bolts (DB)	B.8b	SCC	Cracking/ Fracture	Localized degradation of structural integrity	No operational effects. Possibly minor damage to the thermal shield.	C	1	2	No.	

Failure Modes, Effects, and Criticality Analysis (FMECA) Table

Component Item Name	FMECA Identifier	Degradation Mechanism	Failure Mode	Failure Effect (Local)	Failure Effect (Global)	Susceptibility	Severity of Consequences (Safety)	Severity of Consequences (Economic)	Detectable	Comments
		Fatigue (Thermal Stress Relaxation/ Creep)	Cracking/ Fracture	Loss of mechanical joint integrity	No operational effects. Possibly minor damage to the thermal shield. If locking devices fail (B.9 has no credible degradation mechanism identified), broken pieces from locking devices and/or bolts could result in loose parts. Also applies to wear.	B	1	2	No.	
		Wear (Thermal Stress Relaxation/Creep)	Loss of material	Loss of mechanical joint integrity	No operational effects. Possibly minor damage to the thermal shield.	B	1	2	No.	
Replacement Surveillance Specimen Holder Tube-to-Thermal Shield Bolts Locking Cups & Tie Plates (CR-3, DB)	B.9	No credible degradation mechanism	n/a	n/a	n/a	A	Not evaluated	Not evaluated		The screening analysis shows that none of the screening parameter values are exceeded (Category A) and therefore the likelihood of any consequential age-related degradation is extremely small. Accordingly, this component item is only evaluated (and screened) on a susceptibility basis. (No evaluation of consequence is performed.)
Thermal Shield Cylinder	B.10	SCC	Cracking/ Fracture	Localized degradation of structural integrity (at the HAZ)	No operational effects.	B	1	2	No.	Not in the Generic License Renewal Scope. SCC cracking is tight. Does not permit any significant flow diversion. SCC cracking potential at the heat affected zone (HAZ)
		Wear	Loss of material	Wearing of the top of the thermal shield due to thermal expansion motion (against the "A" and "B" restraint blocks).	No operational effects. Thermal shield may have slightly more "play" in the restraint blocks.	B	1	3	Yes. Neutron noise detection.	Restraint blocks have hardfacing to eliminate wear as a concern for those component items.
		Irradiation Embrittlement	Less flaw tolerant	More susceptible to crack growth when subjected to an external load with a flaw	No operational effects.	B	1	2	No.	More susceptible to crack growth from an external load/force. Decrease in toughness.
Thermal Shield Restraint "A" and "B" Blocks	B.11	No credible degradation mechanism	n/a	n/a	n/a	A	Not evaluated	Not evaluated		Not in the Generic License Renewal Scope. The screening analysis shows that none of the screening parameter values are exceeded (Category A) and therefore the likelihood of any consequential age-related degradation is extremely small. Accordingly, this component item is only evaluated (and screened) on a susceptibility basis. (No evaluation of consequence is performed.)

Component Item Name	FMECA Identifier	Degradation Mechanism	Failure Mode	Failure Effect (Local)	Failure Effect (Global)	Susceptibility	Severity of Consequences (Safety)	Severity of Consequences (Economic)	Detectable	Comments
Thermal Shield Shims	B.12	No credible degradation mechanism	n/a	n/a	n/a	A	Not evaluated	Not evaluated		Not in the Generic License Renewal Scope. The screening analysis shows that none of the screening parameter values are exceeded (Category A) and therefore the likelihood of any consequential age-related degradation is extremely small. Accordingly, this component item is only evaluated (and screened) on a susceptibility basis. (No evaluation of consequence is performed.)
Thermal Shield Restraint Hardfacing	B.13	No credible degradation mechanism	n/a	n/a	n/a	A	Not evaluated	Not evaluated		Not in the Generic License Renewal Scope. The screening analysis shows that none of the screening parameter values are exceeded (Category A) and therefore the likelihood of any consequential age-related degradation is extremely small. Accordingly, this component item is only evaluated (and screened) on a susceptibility basis. (No evaluation of consequence is performed.)
Thermal Shield Plugs	B.14	No credible degradation mechanism	n/a	n/a	n/a	A	Not evaluated	Not evaluated		Not in the Generic License Renewal Scope. The screening analysis shows that none of the screening parameter values are exceeded (Category A) and therefore the likelihood of any consequential age-related degradation is extremely small. Accordingly, this component item is only evaluated (and screened) on a susceptibility basis. (No evaluation of consequence is performed.)
Thermal Shield Dowels	B.15	No credible degradation mechanism	n/a	n/a	n/a	A	Not evaluated	Not evaluated		Not in the Generic License Renewal Scope. The screening analysis shows that none of the screening parameter values are exceeded (Category A) and therefore the likelihood of any consequential age-related degradation is extremely small. Accordingly, this component item is only evaluated (and screened) on a susceptibility basis. (No evaluation of consequence is performed.)
Thermal Shield Cap Screws	B.16	Fatigue (Thermal and Irradiation Stress Relaxation/Creep)	Cracking/ Fracture	Loss of mechanical joint integrity	No operational effects.	C	1	2	No.	
		Wear (Thermal and Irradiation Stress Relaxation/Creep)	Loss of material	Loss of mechanical joint integrity	No operational effects.	C	1	2	No.	
Baffle Plates	B.17	IASCC	Cracking/ Fracture	Localized degradation of structural integrity	No operational effects. May be minor fuel damage from interaction between baffle plate and outer fuel assembly.	C	2	2	Yes. If fuel failure, fuel assemblies will be examined during refuel outage.	SCC cracking is tight. Does not permit any significant flow diversion. Water jetting is not a concern.
		Irradiation Embrittlement	Less flaw tolerant	More susceptible to crack growth when subjected to an external load with a flaw	No operational effects.	D	1	2	No.	More susceptible to crack growth from an external load/force. Decrease in toughness.

Failure Modes, Effects, and Criticality Analysis (FMECA) Table

Component Item Name	FMECA Identifier	Degradation Mechanism	Failure Mode	Failure Effect (Local)	Failure Effect (Global)	Susceptibility	Severity of Consequences (Safety)	Severity of Consequences (Economic)	Detectable	Comments
		Void Swelling	Localized embrittleme nt	Localized cracking, very small displacement	No operational effects. May be minor fuel damage from interaction between baffle plate and outer fuel assemblies.	B	2	2	Yes. If fuel failure, fuel assemblies will be examined during refuel outage.	Void swelling below the critical percentage will have minimal impact on the internals. Since void swelling is a temporal function of temperature and neutron fluence, it is not credible that all items will be affected the same at the same time. Further, it is expected that the most significant void swelling effects will occur at the end of plant life. Severe void swelling will have a significant impact on embrittlement; however, this level of swelling is not expected during the life of B&W-designed plants.
Former Plates	B.18	IASCC	Cracking/ Fracture	Localized degradation of structural integrity	No operational effects. May be minor fuel damage from interaction between baffle plate and outer fuel assemblies.	C	2	2	Yes. If fuel failure, fuel assemblies will be examined during refuel outage.	
		Irradiation Embrittlement	Less flaw tolerant	More susceptible to crack growth when subjected to an external load with a flaw	No operational effects.	D	1	2	No.	More susceptible to crack growth from an external load/force. Decrease in toughness.
		Void Swelling	Localized embrittleme nt	Localized cracking, very small displacement	No operational effects. May be minor fuel damage from interaction between baffle plate and outer fuel assemblies.	C	2	2	Yes. If fuel failure, fuel assemblies will be examined during refuel outage.	Void swelling below the critical percentage will have minimal impact on the internals. Since void swelling is a temporal function of temperature and neutron fluence, it is not credible that all components will be affected the same at the same time. Further, it is expected that the most significant void swelling effects will occur at the end of plant life. Severe void swelling will have a significant impact on embrittlement; however, this level of swelling is not expected during the life of B&W-designed plants.
Core Barrel-to-Former Plate Cap Screws	B.19	IASCC	Cracking/ Fracture	Localized degradation of structural integrity	No operational effects. May be minor fuel damage from interaction between baffle plate and outer fuel assemblies.	C	2	2	Yes. If fuel failure, fuel assemblies will be examined during refuel outage.	
		Fatigue (Thermal and Irradiation Stress Relaxation/Creep)	Cracking/ Fracture	Loss of mechanical joint integrity	No operational effects. May be minor fuel damage from interaction between baffle plate and outer fuel assemblies.	C	2	2	Yes. If fuel failure, fuel assemblies will be examined during refuel outage.	
		Irradiation Embrittlement	Less flaw tolerant	More susceptible to crack growth when subjected to an external load with a flaw	No operational effects.	D	1	2	No.	More susceptible to crack growth from an external load/force. Decrease in toughness.
		Void Swelling	Localized embrittleme nt	Localized cracking, very small displacement	No operational effects. May be minor fuel damage from interaction between baffle plate and outer fuel assemblies.	B	2	2	Yes. If fuel failure, fuel assemblies will be examined during refuel outage..	Void swelling below the critical percentage will have minimal impact on the internals. Since void swelling is a temporal function of temperature and neutron fluence, it is not credible that all component items will be affected the same at the same time. Further, it is expected that the most significant void swelling effects will occur at the end of plant life. Severe void swelling will have a significant impact on embrittlement; however, this level of swelling is not expected during the life of B&W-designed plants.
		Wear (Thermal and Irradiation Stress Relaxation/Creep)	Loss of material	Loss of mechanical joint integrity	No operational effects. May be minor fuel damage from interaction between baffle plate and outer fuel assemblies.	C	2	2	Yes. If fuel failure, fuel assemblies will be examined during refuel outage.	

Component Item Name	FMECA Identifier	Degradation Mechanism	Failure Mode	Failure Effect (Local)	Failure Effect (Global)	Susceptibility	Severity of Consequences (Safety)	Severity of Consequences (Economic)	Detectable	Comments
Locking Pins	B.20	IASCC	Cracking/ Fracture	Localized degradation of structural integrity	Broken pieces from locking pins could result in loose parts. Could allow cap screw to come loose (work their way out) leading to loss of pre-load resulting in the same affect as item B.18.	C	1	1	No.	
		Irradiation Embrittlement	Less flaw tolerant	More susceptible to crack growth when subjected to an external load with a flaw	No operational effects.	D	1	1	No.	More susceptible to crack growth from an external load/force. Decrease in toughness.
		Void Swelling	Localized embrittleme nt	Localized cracking, very small displacement	No operational effects.	B	1	1	No.	Void swelling below the critical percentage will have minimal impact on the internals. Since void swelling is a temporal function of temperature and neutron fluence, it is not credible that all component items will be affected the same at the same time. Further, it is expected that the most significant void swelling effects will occur at the end of plant life. Severe void swelling will have a significant impact on embrittlement; however, this level of swelling is not expected during the life of B&W-designed plants.
Dowels	B.21	Irradiation Embrittlement	Less flaw tolerant	More susceptible to crack growth when subjected to an external load with a flaw	No operational effects.	D	1	1	No.	Dowels used for alignment during assembly. More susceptible to crack growth from an external load/force. Decrease in toughness.
		Void Swelling	Localized embrittleme nt	Localized cracking, very small displacement	No operational effects.	B	1	1	No.	Void swelling below the critical percentage will have minimal impact on the internals. Since void swelling is a temporal function of temperature and neutron fluence, it is not credible that all component items will be affected the same at the same time. Further, it is expected that the most significant void swelling effects will occur at the end of plant life. Severe void swelling will have a significant impact on embrittlement; however, this level of swelling is not expected during the life of B&W-designed plants.
Baffle Plates-to-Former Plates Bolts	B.22	Fatigue (Thermal and Irradiation Stress Relaxation/Creep)	Cracking/ Fracture	Loss of mechanical joint integrity	No operational effects. May be minor fuel damage from interaction between baffle plate and outer fuel assemblies.	C	2	2	Yes. If fuel failure, fuel assemblies will be examined during refuel outage.	
		IASCC	Cracking/ Fracture	Localized degradation of structural integrity	No operational effects. May be minor fuel damage from interaction between baffle plate and outer fuel assemblies.	C	2	2	Yes. If fuel failure, fuel assemblies will be examined during refuel outage.	
		Irradiation Embrittlement	Less flaw tolerant	More susceptible to crack growth when subjected to an external load with a flaw	No operational effects.	D	1	2	No.	More susceptible to crack growth from an external load/force. Decrease in toughness.

Failure Modes, Effects, and Criticality Analysis (FMECA) Table

Component Item Name	FMECA Identifier	Degradation Mechanism	Failure Mode	Failure Effect (Local)	Failure Effect (Global)	Susceptibility	Severity of Consequences (Safety)	Severity of Consequences (Economic)	Detectable	Comments
		Void Swelling	Localized embrittlement	Localized cracking, very small displacement	No operational effects. May be minor fuel damage from interaction between baffle plate and outer fuel assemblies.	B	2	2	Yes. If fuel failure, fuel assemblies will be examined during refuel outage.	Void swelling below the critical percentage will have minimal impact on the internals. Since void swelling is a temporal function of temperature and neutron fluence, it is not credible that all component items will be affected the same at the same time. Further, it is expected that the most significant void swelling effects will occur at the end of plant life. Severe void swelling will have a significant impact on embrittlement; however, this level of swelling is not expected during the life of B&W-designed plants.
		Wear (Thermal and Irradiation Stress Relaxation/Creep)	Loss of material	Loss of mechanical joint integrity	No operational effects. May be minor fuel damage from interaction between baffle plate and outer fuel assemblies.	C	2	2	Yes. If fuel failure, fuel assemblies will be examined during refuel outage.	
Locking Pins	B.23	IASCC	Cracking/ Fracture	Localized degradation of structural integrity	Broken pieces from locking pins could result in loose parts. Could allow bolt to come loose (work their way out) leading to loss of pre-load resulting in the same affect as item B.22.	C	1	1	No.	
		Irradiation Embrittlement	Less flaw tolerant	More susceptible to crack growth when subjected to an external load with a flaw	No operational effects.	D	1	1	No.	More susceptible to crack growth from an external load/force. Decrease in toughness.
		Void Swelling	Localized embrittlement	Localized cracking, very small displacement	No operational effects.	B	1	1	No.	Void swelling below the critical percentage will have minimal impact on the internals. Since void swelling is a temporal function of temperature and neutron fluence, it is not credible that all component items will be affected the same at the same time. Further, it is expected that the most significant void swelling effects will occur at the end of plant life. Severe void swelling will have a significant impact on embrittlement; however, this level of swelling is not expected during the life of B&W-designed plants.
Baffle Plates-to-Former Plates Shoulder Cap Screws	B.24	IASCC	Cracking/ Fracture	Localized degradation of structural integrity	No operational effects. May be minor fuel damage from interaction between baffle plate and outer fuel assemblies.	C	2	2	Yes. If fuel failure, fuel assemblies will be examined during refuel outage.	These screws are in the fourth former level, and are different from the baffle plate-to-former plate bolts (item B.22) because of a gap between the former plates and the baffle plates. These bolts (alone) are sufficient to hold the baffle plates in place (during a LOCA).
		Fatigue (Thermal and Irradiation Stress Relaxation/Creep)	Cracking/ Fracture	Loss of mechanical joint integrity	No operational effects. May be minor fuel damage from interaction between baffle plate and outer fuel assemblies.	C	2	2	Yes. If fuel failure, fuel assemblies will be examined during refuel outage.	These screws are in the fourth former level, and are different from the baffle plate-to-former plate bolts (item B.22) because of a gap between the former plates and the baffle plates. These bolts (alone) are sufficient to hold the baffle plates in place (during a LOCA).

Component Item Name	FMECA Identifier	Degradation Mechanism	Failure Mode	Failure Effect (Local)	Failure Effect (Global)	Susceptibility	Severity of Consequences (Safety)	Severity of Consequences (Economic)	Detectable	Comments
		Irradiation Embrittlement	Less flaw tolerant	More susceptible to crack growth when subjected to an external load with a flaw	No operational effects.	D	1	2	No.	These screws are in the fourth former level, and are different from the baffle plate-to-former plate bolts (item B.22) because of a gap between the former plates and the baffle plates. These bolts (alone) are sufficient to hold the baffle plates in place (during a LOCA). More susceptible to crack growth from an external load/force. Decrease in toughness.
		Void Swelling	Localized embrittlement	Localized cracking, very small displacement	No operational effects. May be minor fuel damage from interaction between baffle plate and outer fuel assemblies.	B	2	2	Yes. If fuel failure, fuel assemblies will be examined during refuel outage.	Void swelling below the critical percentage will have minimal impact on the internals. Since void swelling is a temporal function of temperature and neutron fluence, it is not credible that all component items will be affected the same at the same time. Further, it is expected that the most significant void swelling effects will occur at the end of plant life. Severe void swelling will have a significant impact on embrittlement; however, this level of swelling is not expected during the life of B&W-designed plants.
		Wear (Thermal and Irradiation Stress Relaxation/Creep)	Loss of material	Loss of mechanical joint integrity	No operational effects. May be minor fuel damage from interaction between baffle plate and outer fuel assemblies.	C	2	2	Yes. If fuel failure, fuel assemblies will be examined during refuel outage.	These screws are in the fourth former level, and are different from the baffle plate-to-former plate bolts (item B.22) because of a gap between the former plates and the baffle plates. These bolts (alone) are sufficient to hold the baffle plates in place (during a LOCA).
Locking Dowels	B.25	IASCC	Cracking/ Fracture	Localized degradation of structural integrity	Broken pieces from locking dowels and/or broken shoulder screws could result in loose parts. Could allow shoulder screws to come loose (work their way out) leading to possible loss of pre-load resulting in the same affect as item B.24.	C	1	1	No.	
		Irradiation Embrittlement	Less flaw tolerant	More susceptible to crack growth when subjected to an external load with a flaw	No operational effects.	D	1	1	No.	More susceptible to crack growth from an external load/force. Decrease in toughness.
		Void Swelling	Localized embrittlement	Localized cracking, very small displacement	No operational effects.	B	1	1	No.	Void swelling below the critical percentage will have minimal impact on the internals. Since void swelling is a temporal function of temperature and neutron fluence, it is not credible that all component items will be affected the same at the same time. Further, it is expected that the most significant void swelling effects will occur at the end of plant life. Severe void swelling will have a significant impact on embrittlement; however, this level of swelling is not expected during the life of B&W-designed plants.
Baffle Plates-to-Baffle Plates Bolts	B.26	IASCC	Cracking/ Fracture	Localized degradation of structural integrity	No operational effects. May be minor fuel damage from interaction between baffle plate and outer fuel assemblies.	C	2	2	Yes. If fuel failure, fuel assemblies will be examined during refuel outage.	

Failure Modes, Effects, and Criticality Analysis (FMECA) Table

Component Item Name	FMECA Identifier	Degradation Mechanism	Failure Mode	Failure Effect (Local)	Failure Effect (Global)	Susceptibility	Severity of Consequences (Safety)	Severity of Consequences (Economic)	Detectable	Comments
		Fatigue (Thermal and Irradiation Stress Relaxation/Creep)	Cracking/ Fracture	Loss of mechanical joint integrity	No operational effects. May be minor fuel damage from interaction between baffle plate and outer fuel assemblies.	C	2	2	Yes. If fuel failure, fuel assemblies will be examined during refuel outage.	
		Irradiation Embrittlement	Less flaw tolerant	More susceptible to crack growth when subjected to an external load with a flaw	No operational effects.	D	1	2	No.	More susceptible to crack growth from an external load/force. Decrease in toughness.
		Void Swelling	Localized embrittleme nt	Localized cracking, very small displacement	No operational effects. May be minor fuel damage from interaction between baffle plate and outer fuel assemblies.	B	2	2	Yes. If fuel failure, fuel assemblies will be examined during refuel outage.	Void swelling below the critical percentage will have minimal impact on the internals. Since void swelling is a temporal function of temperature and neutron fluence, it is not credible that all component items will be affected the same at the same time. Further, it is expected that the most significant void swelling effects will occur at the end of plant life. Severe void swelling will have a significant impact on embrittlement; however, this level of swelling is not expected during the life of B&W-designed plants.
		Wear (Thermal and Irradiation Stress Relaxation/Creep)	Loss of material	Loss of mechanical joint integrity	No operational effects. May be minor fuel damage from interaction between baffle plate and outer fuel assemblies.	C	2	2	Yes. If fuel failure, fuel assemblies will be examined during refuel outage.	
Locking Rings	B.27	IASCC	Cracking/ Fracture	Localized degradation of structural integrity	Broken pieces from locking rings could result in loose parts. Could allow bolt to come loose (work their way out) leading to loss of pre-load resulting in the same affect as item B.26.	C	1	1	No.	
		Irradiation Embrittlement	Less flaw tolerant	More susceptible to crack growth when subjected to an external load with a flaw	No operational effects.	D	1	1	No.	More susceptible to crack growth from an external load/force. Decrease in toughness.
		Void Swelling	Localized embrittleme nt	Localized cracking, very small displacement	No operational effects.	B	1	1	No.	Void swelling below the critical percentage will have minimal impact on the internals. Since void swelling is a temporal function of temperature and neutron fluence, it is not credible that all component items will be affected the same at the same time. Further, it is expected that the most significant void swelling effects will occur at the end of plant life. Severe void swelling will have a significant impact on embrittlement; however, this level of swelling is not expected during the life of B&W-designed plants.

Table A-4
Lower Internals Assembly (L)

Component Item Name	FMECA Identifier	Degradation Mechanism	Failure Mode	Failure Effect (Local)	Failure Effect (Global)	Likelihood of Occurrence	Severity of Consequences (Safety)	Severity of Consequences (Economic)	Detectable	Comments
Lower Grid Assembly	L.1	Function	Provides alignment and support for fuel assemblies Supports the core barrel assembly and flow distributor Aligns the IMI guide tubes (with fuel assembly instrument tubes)							
Rib Section	L.1.1	Irradiation Embrittlement	Less flaw tolerant	More susceptible to crack growth when subjected to an external load with a flaw	No operational effects.	B	1	2	No.	More susceptible to crack growth from an external load/force. Decrease in toughness. Economic consequences driven by analysis and/or repairs.
Fuel Assembly Support Pads	L.1.2	Irradiation Embrittlement	Less flaw tolerant	More susceptible to crack growth when subjected to an external load with a flaw	No operational effects.	C	1	1	No.	More susceptible to crack growth from an external load/force. Decrease in toughness.
Dowels	L.1.3	Irradiation Embrittlement	Less flaw tolerant	More susceptible to crack growth when subjected to an external load with a flaw	No operational effects.	C	1	1	No.	More susceptible to crack growth from an external load/force. Decrease in toughness.
Cap Screws	L.1.4	Fatigue (Thermal and Irradiation Stress Relaxation/Creep)	Cracking/ Fracture	Loss of mechanical joint integrity	No operational effects. Cracking around the weld could allow cap screw to become "unlocked" – no effect unless there is also loss of pre-load.	C	1	2	No.	Economic consequences driven by analysis and/or repairs.
		Irradiation Embrittlement	Less flaw tolerant	More susceptible to crack growth when subjected to an external load with a flaw	No operational effect	C	1	1	No.	More susceptible to crack growth from an external load/force. Decrease in toughness.
		Wear (Thermal and Irradiation Stress Relaxation/Creep)	Loss of material	Loss of mechanical joint integrity	No operational effects, fuel assembly support pad held in place by two dowels and fuel assemblies on either side.	C	1	2	No.	Economic consequences driven by analysis and/or repair.
Rib Section-to-Shell Forging Cap Screws	L.1.5	Fatigue (Thermal and Irradiation Stress Relaxation/Creep)	Cracking/ Fracture	Loss of mechanical joint integrity	No operational effects. Redundancy in configuration and welded support pipes will maintain the integrity of the lower grid assembly. If locking pins fail (L.1.6 has no credible degradation mechanism identified), broken pieces from locking pins and/or bolts could result in loose parts. Also applies to wear.	C	1	2	No.	Economic consequences driven by analysis and/or repairs.

Component Item Name	FMECA Identifier	Degradation Mechanism	Failure Mode	Failure Effect (Local)	Failure Effect (Global)	Likelihood of Occurrence	Severity of Consequences (Safety)	Severity of Consequences (Economic)	Detectable	Comments
		Irradiation Embrittlement	Less flaw tolerant	More susceptible to crack growth when subjected to an external load with a flaw	No operational effects.	C	1	2	No.	More susceptible to crack growth from an external load/force. Decrease in toughness.
		Wear (Thermal and Irradiation Stress Relaxation/Creep)	Loss of material	Loss of mechanical joint integrity	No operational effects. Redundancy in configuration and welded support pipes will maintain the integrity of the lower grid assembly.	C	1	2	No.	Economic consequences driven by analysis and/or repair.
Rib Section-to-Shell Forging Cap Screw Locking Pins	L.1.6	No credible degradation mechanism	n/a	n/a	n/a	A	Not evaluated	Not evaluated		The screening analysis shows that none of the screening parameter values are exceeded (Category A) and therefore the likelihood of any consequential age-related degradation is extremely small. Accordingly, this component item is only evaluated (and screened) on a susceptibility basis. (No evaluation of consequence is performed.)
Lower Grid Flow Distributor Plate	L.1.7	SCC	Cracking/ Fracture	Localized degradation of structural integrity (at the HAZ)	Cracks around the weld between the lower grid distributor plate and the lower grid forging. No operational effects as the distributor plate is held firmly in place between the lower grid rib section and the lower grid forging.	B	1	2	No.	SCC cracking is tight. Does not permit any significant flow diversion. Economic consequences driven by analysis and/or repair.
		Fatigue	Cracking/ Fracture	Localized degradation of structural integrity (at the HAZ)	No operational effects as the distributor plate is held firmly in place between the lower grid rib section and the lower grid forging.	B	1	2	No.	Economic consequences driven by analysis and/or repair.
		Irradiation Embrittlement	Less flaw tolerant	More susceptible to crack growth when subjected to an external load with a flaw	No operational effects.	B	1	2	No.	More susceptible to crack growth from an external load/force. Decrease in toughness. Less influence on embrittlement from the fluence at this location.
Orifice Plugs	L.1.8	SCC	Cracking/ Fracture	Localized degradation of structural integrity (at the HAZ)	Broken pieces from orifice plugs could result in loose parts.	B	1	1	No.	Loose parts probably will not travel and generate enough energy to "make noise."
		Irradiation Embrittlement	Less flaw tolerant	More susceptible to crack growth when subjected to an external load with a flaw	No operational effects.	B	1	1	No.	More susceptible to crack growth from an external load/force. Decrease in toughness. Less influence on embrittlement from the fluence at this location.
Lower Grid Forging (ANO-1, CR-3, DB, ONS-2,3, TMI-1)	L.1.9a	SCC	Cracking/ Fracture	Localized degradation of structural integrity (at the HAZ)	Cracks around the weld between the lower grid forging and the lower grid shell forging. No operational effects.	B	1	2	No.	SCC cracking is tight. Does not permit any significant flow diversion. Economic consequences driven by analysis and/or repair.
		Fatigue	Cracking/ Fracture	Localized degradation of structural integrity (at the HAZ)	No operational effects.	B	1	2	No.	Economic consequences driven by analysis and/or repair.

Component Item Name	FMECA Identifier	Degradation Mechanism	Failure Mode	Failure Effect (Local)	Failure Effect (Global)	Likelihood of Occurrence	Severity of Consequences (Safety)	Severity of Consequences (Economic)	Detectable	Comments
Lower Grid Weldment Ribs (ONS-1)	L.1.9b	SCC	Cracking/ Fracture	Localized degradation of structural integrity (at the HAZ)	Cracks around the weld between the lower grid weldment ribs and the lower grid shell forging. No operational effects.	B	1	2	No.	SCC cracking is tight. Does not permit any significant flow diversion. Economic consequences driven by analysis and/or repair.
		Fatigue	Cracking/ Fracture	Localized degradation of structural integrity (at the HAZ)	No operational effects.	B	1	2	No.	Economic consequences driven by analysis and/or repair.
Lower Grid Shell Forging	L.1.10	SCC	Cracking/ Fracture	Localized degradation of structural integrity (at the HAZ)	Cracks around the weld between the lower grid forging and the lower grid shell forging. No operational effects.	B	1	2	No.	Economic consequences driven by analysis and/or repair.
Original Lower Grid Assembly-to-Thermal Shield Bolts (TMI-1)	L.1.11a	SCC	Cracking/ Fracture	Localized degradation of structural integrity	Possible vibrations in thermal shield could cause fatigue of the thermal shield. Thermal shield is shrunk-fit and bolted to lower grid shell forging, and restrained at the top.	C	1	2	No.	Economic consequences driven by analysis and/or repair.
		Fatigue (Thermal Stress Relaxation/ Creep)	Cracking/ Fracture	Loss of mechanical joint integrity	Possible vibrations in thermal shield could cause fatigue of the thermal shield. Thermal shield is shrunk-fit and bolted to lower grid shell forging, and restrained at the top. If locking devices fail (L.1.12 has no credible degradation mechanism identified), broken pieces from locking devices and/or bolts could result in loose parts. Also applies to wear.	B	1	2	No.	Economic consequences driven by analysis and/or repair.
		Wear (Thermal Stress Relaxation/Creep)	Loss of material	Loss of mechanical joint integrity	Possible vibrations in thermal shield could cause fatigue of the thermal shield. Thermal shield is shrunk-fit and bolted to lower grid shell forging, and restrained at the top.	B	1	2	No.	Economic consequences driven by analysis and/or repair.

Failure Modes, Effects, and Criticality Analysis (FMECA) Table

Component Item Name	FMECA Identifier	Degradation Mechanism	Failure Mode	Failure Effect (Local)	Failure Effect (Global)	Likelihood of Occurrence	Severity of Consequences (Safety)	Severity of Consequences (Economic)	Detectable	Comments
Replacement Lower Grid Assembly-to-Thermal Shield Studs and Nuts (CR-3, ONS)	L.1.11b	SCC	Cracking/ Fracture	Localized degradation of structural integrity	Possible vibrations in thermal shield could cause fatigue of the thermal shield. Thermal shield is shrunk-fit and bolted to lower grid shell forging, and restrained at the top.	C	1	2	No.	Economic consequences driven by analysis and/or repair.
		Fatigue (Thermal Stress Relaxation/ Creep)	Cracking/ Fracture	Loss of mechanical joint integrity	Possible vibrations in thermal shield could cause fatigue of the thermal shield. Thermal shield is shrunk-fit and bolted to lower grid shell forging, and restrained at the top. If locking devices fail (L.1.12 has no credible degradation mechanism identified), broken pieces from locking devices and/or bolts could result in loose parts. Also applies to wear.	B	1	2	No.	Economic consequences driven by analysis and/or repair.
		Wear (Thermal Stress Relaxation/Creep)	Loss of material	Loss of mechanical joint integrity	Possible vibrations in thermal shield could cause fatigue of the thermal shield. Thermal shield is shrunk-fit and bolted to lower grid shell forging, and restrained at the top.	B	1	2	No.	Economic consequences driven by analysis and/or repair.
Replacement Lower Grid Assembly-to-Thermal Shield Bolts (DB, ANO-1)	L.1.11c	SCC	Cracking/ Fracture	Localized degradation of structural integrity	Possible vibrations in thermal shield could cause fatigue of the thermal shield. Thermal shield is shrunk-fit and bolted to lower grid shell forging, and restrained at the top.	C	1	2	No.	Economic consequences driven by analysis and/or repair.

Component Item Name	FMECA Identifier	Degradation Mechanism	Failure Mode	Failure Effect (Local)	Failure Effect (Global)	Likelihood of Occurrence	Severity of Consequences (Safety)	Severity of Consequences (Economic)	Detectable	Comments
		Fatigue (Thermal Stress Relaxation/ Creep)	Cracking/ Fracture	Loss of mechanical joint integrity	Possible vibrations in thermal shield could cause fatigue of the thermal shield. Thermal shield is shrunk-fit and bolted to lower grid shell forging, and restrained at the top. If locking devices fail (L.1.12 has no credible degradation mechanism identified), broken pieces from locking devices and/or bolts could result in loose parts. Also applies to wear.	B	1	2	No.	Economic consequences driven by analysis and/or repair.
		Wear (Thermal Stress Relaxation/Creep)	Loss of material	Loss of mechanical joint integrity	Possible vibrations in thermal shield could cause fatigue of the thermal shield. Thermal shield is shrunk-fit and bolted to lower grid shell forging, and restrained at the top.	B	1	2	No.	Economic consequences driven by analysis and/or repair.
Replacement Lower Grid Assembly-to-Thermal Shield Bolts/Studs and Nuts Locking Cups & Tie Plates (ANO-1, CR-3, DB, ONS, TMI-1)	L.1.12	No credible degradation mechanism	n/a	n/a	n/a	A	Not evaluated	Not evaluated		The screening analysis shows that none of the screening parameter values are exceeded (Category A) and therefore the likelihood of any consequential age-related degradation is extremely small. Accordingly, this component item is only evaluated (and screened) on a susceptibility basis. (No evaluation of consequence is performed.)
Guide Blocks	L.1.13	SCC	Cracking/ Fracture	Localized degradation of structural integrity (at the HAZ)	Cracking around bolt fillet weld and/or dowel fillet weld. No operational effects. Bolt will hold the guide block in place.	B	1	2	No.	Economic consequences driven by analysis and/or repair.
		Wear	Loss of material	Increase the gap between the guide block(s) and the associated guide lug.	Possibly increase the motion of the internals; increased vibration.	B	1	2	No.	Economic consequences driven by analysis and/or repair. Wear could be caused by installation of internals or loss of clamp-up (see P.1.2 and P.1.4).
Guide Block Bolts	L.1.14a	Fatigue (Thermal Stress Relaxation/ Creep)	Cracking/ Fracture	Loss of mechanical joint integrity	No operational effects. While a bolt may fail, the guide block will remain in place since there will be a guide lug on one side. May affect the ability to insert internals into the reactor vessel.	B	1	2	Yes, if insertion of internals are affected.	Economic consequences driven by analysis and/or repair.

Component Item Name	FMECA Identifier	Degradation Mechanism	Failure Mode	Failure Effect (Local)	Failure Effect (Global)	Likelihood of Occurrence	Severity of Consequences (Safety)	Severity of Consequences (Economic)	Detectable	Comments
		Wear (Thermal Stress Relaxation/Creep)	Loss of material	Loss of mechanical joint integrity	No operational effects. While a bolt may fail, the guide block will remain in place since there will be a guide lug on one side. May affect the ability to insert internals into the reactor vessel.	B	1	2	Yes, if insertion of internals are affected.	Economic consequences driven by analysis and/or repair.
Guide Block Washers	L.1.14b	No credible degradation mechanism	n/a	n/a	n/a	A	Not evaluated	Not evaluated		The screening analysis shows that none of the screening parameter values are exceeded (Category A) and therefore the likelihood of any consequential age-related degradation is extremely small. Accordingly, this component item is only evaluated (and screened) on a susceptibility basis. (No evaluation of consequence is performed.)
Dowels	L.1.15	No credible degradation mechanism	n/a	n/a	n/a	A	Not evaluated	Not evaluated		The screening analysis shows that none of the screening parameter values are exceeded (Category A) and therefore the likelihood of any consequential age-related degradation is extremely small. Accordingly, this component item is only evaluated (and screened) on a susceptibility basis. (No evaluation of consequence is performed.)
Shock Pads (ANO-1, CR-3, DB, ONS)	L.1.16a	No credible degradation mechanism	n/a	n/a	n/a	A	Not evaluated	Not evaluated		The screening analysis shows that none of the screening parameter values are exceeded (Category A) and therefore the likelihood of any consequential age-related degradation is extremely small. Accordingly, this component item is only evaluated (and screened) on a susceptibility basis. (No evaluation of consequence is performed.)
Shock Pads (TMI-1)	L.1.16b	No credible degradation mechanism	n/a	n/a	n/a	A	Not evaluated	Not evaluated		The screening analysis shows that none of the screening parameter values are exceeded (Category A) and therefore the likelihood of any consequential age-related degradation is extremely small. Accordingly, this component item is only evaluated (and screened) on a susceptibility basis. (No evaluation of consequence is performed.)
Shock Pads Bolts (ANO-1, CR-3, DB, ONS)	L.1.17a	Fatigue (Thermal Stress Relaxation/ Creep)	Cracking/ Fracture	Loss of mechanical joint integrity	No effect if a small number of bolts are degraded. Each shock pad has two bolts; there are 12 shock pads.	B	1	2	No.	Economic consequences driven by analysis and/or repair.
					If a significant number of bolts were degraded, most or all of shock pads could be affected. The guide lugs are designed for the shock pads to drop onto them. Loss of most or all shock pads would leave a 2½" gap should the internals drop.	**B/I**	**Not evaluated**	**Not evaluated**		Failed shock pads are only a concern if some portion of the internals drops. The shock pads limit the drop to the guide lugs to about a ½ inch, thus ensuring continued efficacy of the control rod drives. However, note, that while not in the PWR internals scope, the guide lugs need to maintain functionality to stop the fall of the internals. The loss of the shock pads does not make the dropping of the internals more likely (so this is not a dependent or cascading failure). However, the consequences of not having the shock pads in place if the internals drop are potentially catastrophic.

Component Item Name	FMECA Identifier	Degradation Mechanism	Failure Mode	Failure Effect (Local)	Failure Effect (Global)	Likelihood of Occurrence	Severity of Consequences (Safety)	Severity of Consequences (Economic)	Detectable	Comments
		Wear (Thermal Stress Relaxation/Creep)	Loss of material	Loss of mechanical joint integrity	No effect if a small number of bolts are degraded. Each shock pad has two bolts; there are 12 shock pads.	B	1	2	No.	Economic consequences driven by analysis and/or repair.
					If a significant number of bolts were degraded, most or all of shock pads could be affected. The guide lugs are designed for the shock pads to drop onto them. Loss of most or all shock pads would leave a 2½" gap should the internals drop.	**B/I**	**Not evaluated**	**Not evaluated**		Failed shock pads are only a concern if some portion of the internals drops. The shock pads limit the drop to the guide lugs to about a ½ inch, thus ensuring continued efficacy of the control rod drives. However, note, that while not in the PWR internals scope, the guide lugs need to maintain functionality to stop the fall of the internals. The loss of the shock pads does not make the dropping of the internals more likely (so this is not a dependent or cascading failure). However, the consequences of not having the shock pads in place if the internals drop are potentially catastrophic.
Shock Pad Bolts (TMI-1)	L.1.17b	SCC	Cracking/ Fracture	Loss of pre-load	No effect if a small number of bolts are degraded. Each shock pad has two bolts; there are 12 shock pads.	C	1	2	No.	Economic consequences driven by analysis and/or repair.
					If a significant number of bolts were degraded, most or all of shock pads could be affected. The guide lugs are designed for the shock pads to drop onto them. Loss of most or all shock pads would leave a 2½" gap should the internals drop.	**C/I**	**Not evaluated**	**Not evaluated**		Failed shock pads are only a concern if some portion of the internals drops. The shock pads limit the drop to the guide lugs to about a ½ inch, thus ensuring continued efficacy of the control rod drives. However, note, that while not in the PWR internals scope, the guide lugs need to maintain functionality to stop the fall of the internals. The loss of the shock pads does not make the dropping of the internals more likely (so this is not a dependent or cascading failure). However, the consequences of not having the shock pads in place if the internals drop are potentially catastrophic.
		Fatigue (Thermal Stress Relaxation/ Creep)	Cracking/ Fracture	Loss of mechanical joint integrity	No effect if a small number of bolts are degraded. Each shock pad has two bolts; there are 12 shock pads.	B	1	2	No.	Economic consequences driven by analysis and/or repair.
					If a significant number of bolts were degraded, most or all of shock pads could be affected. The guide lugs are designed for the shock pads to drop onto them. Loss of most or all shock pads would leave a 2½" gap should the internals drop.	**B/I**	**Not evaluated**	**Not evaluated**		Failed shock pads are only a concern if some portion of the internals drops. The shock pads limit the drop to the guide lugs to about a ½ inch, thus ensuring continued efficacy of the control rod drives. However, note, that while not in the PWR internals scope, the guide lugs need to maintain functionality to stop the fall of the internals. The loss of the shock pads does not make the dropping of the internals more likely (so this is not a dependent or cascading failure). However, the consequences of not having the shock pads in place if the internals drop are potentially catastrophic.

Failure Modes, Effects, and Criticality Analysis (FMECA) Table

Component Item *Name*	FMECA *Identifier*	Degradation *Mechanism*	Failure *Mode*	Failure Effect (Local)	Failure Effect (Global)	Likelihood of *Occurrence*	Severity of Consequences (Safety)	Severity of Consequences (Economic)	Detectable	Comments
		Wear (Thermal Stress Relaxation/Creep)	Loss of material	Loss of mechanical joint integrity	No effect if a small number of bolts are degraded. Each shock pad has two bolts; there are 12 shock pads.	B	1	2	No.	Economic consequences driven by analysis and/or repair.
					If a significant number of bolts were degraded, most or all of shock pads could be affected. The guide lugs are designed for the shock pads to drop onto them. Loss of most or all shock pads would leave a 2½" gap should the internals drop.	**B/I**	**Not evaluated**	**Not evaluated**		Failed shock pads are only a concern if some portion of the internals drops. The shock pads limit the drop to the guide lugs to about a ½ inch, thus ensuring continued efficacy of the control rod drives. However, note, that while not in the PWR internals scope, the guide lugs need to maintain functionality to stop the fall of the internals. The loss of the shock pads does not make the dropping of the internals more likely (so this is not a dependent or cascading failure). However, the consequences of not having the shock pads in place if the internals drop are potentially catastrophic.
Support Post Pipes	L.1.18	SCC	Cracking/ Fracture	Localized degradation of structural integrity (at the HAZ)	Cracking around the welds holding the support pipe to the lower grid forging. No operational effects. Pipe has four notches to allow flow, so cracking will not affect coolant flow.	B	1	2	No. The support posts are not generally accessible for testing.	There are 48 support posts that help to support the lower grid rib assembly. Economic consequences driven by analysis and/or repair.
		Irradiation Embrittlement	Less flaw tolerant	More susceptible to crack growth when subjected to an external load with a flaw	No operational effects.	B	1	2	No.	More susceptible to crack growth from an external load/force. Decrease in toughness. Less influence on embrittlement from the fluence at this location.
Bolting Plugs	L.1.19	SCC	Cracking/ Fracture	Localized degradation of structural integrity (at the HAZ)	Cracking around the welds of the bolting plugs to the support post. No operational effects. Bolting plugs are scalloped to permit flow through the support pipe, so cracking will not affect coolant flow.	B	1	1	No.	
		Irradiation Embrittlement	Less flaw tolerant	More susceptible to crack growth when subjected to an external load with a flaw	No operational effects.	B	1	1	No.	More susceptible to crack growth from an external load/force. Decrease in toughness. Less influence on embrittlement from the fluence at this location.
Support Post Cap Screws	L.1.20	Fatigue (Thermal and Irradiation Stress Relaxation/Creep)	Cracking/ Fracture	Loss of mechanical joint integrity	No operational effects. If locking pins fail (L.1.21 has one credible failure mechanism identified), broken pieces from locking pins and/or screws could result in loose parts. Also applies to wear.	C	1	1	No.	
		Irradiation Embrittlement	Less flaw tolerant	More susceptible to crack growth when subjected to an external load with a flaw	No operational effects.	B	1	1	No.	More susceptible to crack growth from an external load/force. Decrease in toughness. Less influence on embrittlement from the fluence at this location.

Component Item Name	FMECA Identifier	Degradation Mechanism	Failure Mode	Failure Effect (Local)	Failure Effect (Global)	Likelihood of Occurrence	Severity of Consequences (Safety)	Severity of Consequences (Economic)	Detectable	Comments
		Wear (Thermal and Irradiation Stress Relaxation/Creep)	Loss of material	Loss of mechanical joint integrity	No operational effects.	C	1	1	No.	
Locking Pins	L.1.21	Irradiation Embrittlement	Less flaw tolerant	More susceptible to crack growth when subjected to an external load with a flaw	No operational effects.	B	1	1	No.	More susceptible to crack growth from an external load/force. Decrease in toughness. Less influence on embrittlement from the fluence at this location.
Flow Distributor Assembly	L.2	Function	Supports the IMI guide tubes Directs the inlet coolant entering the bottom of the core							
Flow Distributor Head	L.2.1	SCC	Cracking/ Fracture	Localized degradation of structural integrity (at the HAZ)	No operational effects. Critical flaw size (crack) around the weld separating the flow distributor head from the flow distributor flange.	B **B/I**	1 **Not evaluated**	2 **Not evaluated**	No.	The IMI guide tubes, which are welded to the flow distributor head and connected to the IMI guide support plate with a nut would (initially) keep the flow distributor plate from falling. Note that the shock pads are above the bolts that attach the flow distributor flange to the lower grid shell forging – thus, without the IMI guide tubes, the flow distributor head would fall into the bottom of the reactor vessel.
Flow Distributor Flange	L.2.2	SCC	Cracking/ Fracture	Localized degradation of structural integrity (at the HAZ)	No operational effects. Critical flaw size (crack) around the weld separating the flow distributor head from the flow distributor flange.	B **B/I**	1 **Not evaluated**	2 **Not evaluated**	No.	The IMI guide tubes, which are welded to the flow distributor head and connected to the IMI guide support plate with a nut would (initially) keep the flow distributor plate from falling. Note that the shock pads are above the bolts that attach the flow distributor flange to the lower grid shell forging – thus, without the IMI guide tubes, the flow distributor head would fall into the bottom of the reactor vessel.
Flow Distributor-to-Shell Forging Bolts (ANO-1, CR-3, DB, ONS)	L.2.3a	SCC	Cracking/ Fracture	Localized degradation of structural integrity	No effect if a small number of bolts are degraded.	D	1	2	No.	The IMI guide tubes, which are welded to the flow distributor head and connected to the IMI guide support plate with a nut would (initially) keep the flow distributor plate from falling. Note that the shock pads are above the bolts that attach the flow distributor flange to the lower grid shell forging – thus, without the IMI guide tubes, the flow distributor head would fall into the bottom of the reactor vessel. The asterisks (*) indicate that the consequences are as a result of a cascading (or dependent) failure. As some bolts fail, the remaining bolts will be subject to a greater load, which will increase their likelihood of failure.
					If a significant number of bolts were degraded, the flow distributor assembly will be affected. See discussion in comments.	D	3*	4*	Yes. A number of detection means are possible: neutron noises, loss of instru-mentation, loose parts monitoring.	
		Fatigue (Operational Cyclic Loading and	Cracking/ Fracture	Loss of mechanical joint integrity	No effect if a small number of bolts are degraded.	B	1	2	No.	The IMI guide tubes, which are welded to the flow distributor head and connected to the IMI guide support plate with a nut would

Failure Modes, Effects, and Criticality Analysis (FMECA) Table

Component Item Name	FMECA Identifier	Degradation Mechanism	Failure Mode	Failure Effect (Local)	Failure Effect (Global)	Likelihood of Occurrence	Severity of Consequences (Safety)	Severity of Consequences (Economic)	Detectable	Comments
		Thermal Stress Relaxation/Creep)			If a significant number of bolts were degraded, the flow distributor assembly will be affected. See discussion in comments. If locking clips fail (L.2.4 has no credible degradation mechanism identified), broken pieces from locking clips and/or bolts could result in loose parts. Also applies to wear.	B	3*	4*	Yes. A number of detection means are possible: neutron noises, loss of instru-mentation, loose parts monitoring.	(initially) keep the flow distributor plate from falling. Note that the shock pads are above the bolts that attach the flow distributor flange to the lower grid shell forging – thus, without the IMI guide tubes, the flow distributor head would fall into the bottom of the reactor vessel. The asterisks (*) indicate that the consequences are as a result of a cascading (or dependent) failure. As some bolts fail, the remaining bolts will be subject to a greater load, which will increase their likelihood of failure.
		Wear (Thermal Stress Relaxation/Creep)	Loss of material	Loss of mechanical joint integrity	No effect if a small number of bolts are degraded.	B	1	2	No.	The IMI guide tubes, which are welded to the flow distributor head and connected to the IMI guide support plate with a nut would (initially) keep the flow distributor plate from falling. Note that the shock pads are above the bolts that attach the flow distributor flange to the lower grid shell forging – thus, without the IMI guide tubes, the flow distributor head would fall into the bottom of the reactor vessel. The asterisks (*) indicate that the consequences are as a result of a cascading (or dependent) failure. As some bolts fail, the remaining bolts will be subject to a greater load, which will increase their likelihood of failure.
					If a significant number of bolts were degraded, the flow distributor assembly will be affected. See discussion in comments.	B	3*	4*	Yes. A number of detection means are possible: neutron noises, loss of instru-mentation, loose parts monitoring.	
Flow Distributor-to-Shell Forging Bolts (TMI-1)	L.2.3b	SCC	Cracking/ Fracture	Localized degradation of structural integrity	No effect if a small number of bolts are degraded.	D	1	2	No.	The IMI guide tubes, which are welded to the flow distributor head and connected to the IMI guide support plate with a nut would (initially) keep the flow distributor plate from falling. Note that the shock pads are above the bolts that attach the flow distributor flange to the lower grid shell forging – thus, without the IMI guide tubes, the flow distributor head would fall into the bottom of the reactor vessel. The asterisks (*) indicate that the consequences are as a result of a cascading (or dependent) failure. As some bolts fail, the remaining bolts will be subject to a greater load, which will increase their likelihood of failure.
					If a significant number of bolts were degraded, the flow distributor assembly will be affected. See discussion in comments.	D	3*	4*	Yes. A number of detection means are possible: neutron noises, loss of instru-mentation, loose parts monitoring.	
		Fatigue (Operational Cyclic Loading and	Cracking/ Fracture	Loss of mechanical joint integrity	No effect if a small number of bolts are degraded.	B	1	2	No.	The IMI guide tubes, which are welded to the flow distributor head and connected to the IMI guide support plate with a nut would

Component Item Name	FMECA Identifier	Degradation Mechanism	Failure Mode	Failure Effect (Local)	Failure Effect (Global)	Likelihood of Occurrence	Severity of Consequences (Safety)	Severity of Consequences (Economic)	Detectable	Comments
		Thermal Stress Relaxation/Creep)			If a significant number of bolts were degraded, the flow distributor assembly will be affected. See discussion in comments. If locking clips fail (L.2.4 has no credible degradation mechanism identified), broken pieces from locking clips and/or bolts could result in loose parts. Also applies to wear.	B	3*	4*	Yes. A number of detection means are possible: neutron noises, loss of instru-mentation, loose parts monitoring.	(initially) keep the flow distributor plate from falling. Note that the shock pads are above the bolts that attach the flow distributor flange to the lower grid shell forging – thus, without the IMI guide tubes, the flow distributor head would fall into the bottom of the reactor vessel. The asterisks (*) indicate that the consequences are as a result of a cascading (or dependent) failure. As some bolts fail, the remaining bolts will be subject to a greater load, which will increase their likelihood of failure.
		Wear (Thermal Stress Relaxation/Creep)	Loss of material	Loss of mechanical joint integrity	No effect if a small number of bolts are degraded.	B	1	2	No.	The IMI guide tubes, which are welded to the flow distributor head and connected to the IMI guide support plate with a nut would (initially) keep the flow distributor plate from falling. Note that the shock pads are above the bolts that attach the flow distributor flange to the lower grid shell forging – thus, without the IMI guide tubes, the flow distributor head would fall into the bottom of the reactor vessel. The asterisks (*) indicate that the consequences are as a result of a cascading (or dependent) failure. As some bolts fail, the remaining bolts will be subject to a greater load, which will increase their likelihood of failure.
					If a significant number of bolts were degraded, the flow distributor assembly will be affected. See discussion in comments.	B	3*	4*	Yes. A number of detection means are possible: neutron noises, loss of instru-mentation, loose parts monitoring.	
Flow Distributor-to-Shell Forging Bolts Locking Clips (ANO-1, CR-3, DB, ONS, TMI-1)	L.2.4	No credible degradation mechanism	n/a	n/a	n/a	A	Not evaluated	Not evaluated		The screening analysis shows that none of the screening parameter values are exceeded (Category A) and therefore the likelihood of any consequential age-related degradation is extremely small. Accordingly, this component item is only evaluated (and screened) on a susceptibility basis. (No evaluation of consequence is performed.)
IMI Guide Support Plate	L.2.5	No credible degradation mechanism	n/a	n/a	n/a	A	Not evaluated	Not evaluated		The screening analysis shows that none of the screening parameter values are exceeded (Category A) and therefore the likelihood of any consequential age-related degradation is extremely small. Accordingly, this component item is only evaluated (and screened) on a susceptibility basis. (No evaluation of consequence is performed.)
Clamping Ring	L.2.6	SCC	Cracking/ Fracture	Localized degradation of structural integrity (at the HAZ)	No operational effects. Cracking near the weld would not affect the ability of the clamping ring to hold down the IMI Guide Support Plate; even if the weld cracked, the clamping ring is located inside the flow distributor flange and held down by the lower grid shell forging, which are bolted to each other.	B	1	1	No.	

Failure Modes, Effects, and Criticality Analysis (FMECA) Table

Component Item Name	FMECA Identifier	Degradation Mechanism	Failure Mode	Failure Effect (Local)	Failure Effect (Global)	Likelihood of Occurrence	Severity of Consequences (Safety)	Severity of Consequences (Economic)	Detectable	Comments
		Fatigue (Thermal Stress Relaxation/ Creep)	Cracking/ Fracture	Loss of mechanical joint integrity	No operational effects. Even if there was a loss of clamping force, the clamping ring is located inside the flow distributor flange and held down by the lower grid shell forging, which are bolted to each other.	B	1	1	No.	
		Wear (Thermal Stress Relaxation/Creep)	Loss of material	Loss of mechanical joint integrity	No operational effects. Even if there was a loss of clamping force, the clamping ring is located inside the flow distributor flange and held down by the lower grid shell forging, which are bolted to each other.	B	1	1	No.	
Dowel	L.2.7	No credible degradation mechanism	n/a	n/a	n/a	A	Not evaluated	Not evaluated		*The screening analysis shows that none of the screening* parameter values are exceeded (Category A) and therefore the likelihood of any consequential age-related degradation is extremely small. Accordingly, this component item is only evaluated (and screened) on a susceptibility basis. (No evaluation of consequence is performed.)
IMI Guide Tube Assemblies	L.3	Function	Guide the (52) incore monitoring instrumentation (IMI) assemblies from the IMI nozzles in the RV bottom head to the instrument tubes in the fuel assemblies							
IMI Guide Tubes	L.3.1	SCC	Cracking/ Fracture	Localized degradation of structural integrity (at the HAZ)	Cracking at the weld on the flow distributor head or at the gusset welds. (Not all of the IMI guide tubes use gussets.) No operational effects. The guide tubes have an interference fit through the holes of the IMI guide support plate, and are fastened by a guide tube nut and locking clip.	B	1	2	No.	Economic consequences driven by analysis and/or repair.
		Irradiation Embrittlement	Less flaw tolerant	More susceptible to crack growth when subjected to an external load with a flaw	No operational effects.	B	1	2	No.	More susceptible to crack growth from an external load/force. Decrease in toughness. Less influence on embrittlement from the fluence at this location.
Gussets	L.3.2	SCC	Cracking/ Fracture	Localized degradation of structural integrity (at the HAZ)	Cracking at the gusset welds. (Not all of the IMI guide tubes use gussets.) No operational effects. The guide tubes have an interference fit through the holes of the IMI guide support plate, and are fastened by a guide tube nut and locking clip.	B	1	2	No.	Economic consequences driven by analysis and/or repair.

Component Item Name	FMECA Identifier	Degradation Mechanism	Failure Mode	Failure Effect (Local)	Failure Effect (Global)	Likelihood of Occurrence	Severity of Consequences (Safety)	Severity of Consequences (Economic)	Detectable	Comments
Guide Tube Washers	L.3.3	No credible degradation mechanism	n/a	n/a	n/a	A	Not evaluated	Not evaluated		The screening analysis shows that none of the screening parameter values are exceeded (Category A) and therefore the likelihood of any consequential age-related degradation is extremely small. Accordingly, this component item is only evaluated (and screened) on a susceptibility basis. (No evaluation of consequence is performed.)
Guide Tube Nuts	L.3.4	Fatigue (Thermal Stress Relaxation/ Creep)	Cracking/ Fracture	Loss of mechanical joint integrity	No operational effects. The guide tubes have an interference fit through the holes of the IMI guide support plate, are welded to the flow distributor head or gussets, and nuts have locking clips. If locking clips fail (L.3.5 has no credible degradation mechanism identified), broken pieces from locking clips and/or nuts could result in loose parts. Also applies to wear.	B	1	1	No.	
		Wear (Thermal Stress Relaxation/Creep)	Loss of material	Loss of mechanical joint integrity	No operational effects. The guide tubes have an interference fit through the holes of the IMI guide support plate, are welded to the flow distributor head or gussets, and nuts have locking clips.	B	1	1	No.	
Locking Clips	L.3.5	No credible degradation mechanism	n/a	n/a	n/a	A	Not evaluated	Not evaluated		The screening analysis shows that none of the screening parameter values are exceeded (Category A) and therefore the likelihood of any consequential age-related degradation is extremely small. Accordingly, this component item is only evaluated (and screened) on a susceptibility basis. (No evaluation of consequence is performed.)
Spiders	L.3.6	Irradiation Embrittlement	Less flaw tolerant	More susceptible to crack growth when subjected to an external load with a flaw	No operational effects.	C	1	1	No.	More susceptible to crack growth from an external load/force. Decrease in toughness. Note that the IMI guide tube "fits" in the spider assembly and is not physically attached.
		Thermal Aging Embrittlement	Less flaw tolerant	More susceptible to crack growth when subjected to an external load with a flaw	No operational effects.	C	1	1	No.	More susceptible to crack growth from an external load/force. Decrease in toughness. Note that the IMI guide tube "fits" in the spider assembly and is not physically attached.

B
COMPARISON BETWEEN IMT AND FMECA

The following table highlights some of the specific differences in the IMT approach and the FMECA.

IMT	FMECA
Age-related mechanisms identified for component items were "lumped" for degradation effects, e.g., SCC/IASCC/ fatigue leads to cracking.	Age-related mechanisms identified for component items were individually evaluated.
Consequences were identified in broad categories with no explicit severity evaluation.	Local and global consequences were identified for each component item/age-related degradation mechanism. The FMECA provided a semi-quantitative measure of the severity of consequences (using the expert panel). Consequences were evaluated on a safety and economic basis.
There was no explicit susceptibility evaluation of the component item/age-related degradation mechanism pair.	The FMECA provided a semi-quantitative measure of the susceptibility (likelihood of occurrence) of component item/age-related degradation mechanism pair.
There was no ranking metric defined.	The FMECA created enough information to develop a risk-ranking metric by combining the susceptibility with the severity of consequences. Development of a risk matrix lead to the assignment of risk bands from which insight for ranking and categorization can be accomplished.

The assessment of a consequence category in the IMT work (in MRP-156 and MRP-157) was not performed with consideration of the susceptibility of the age-related degradation mechanism. For example, for the core barrel cylinders, MRP-157 lists 10 different age-related degradation mechanisms; the FMECA, drawing from degradation screening criteria results in MRP-175, only identifies two (SCC and fatigue). Furthermore, at the expert panel, the local and global effects for the identified age-related degradation mechanisms were reviewed and adjusted as needed. Accordingly, it is possible to find some differences between the results of the IMT work and what is reflected in the FMECA table.

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