NUCLEAR MANAGEMENT AND RESOURCES COUNCIL

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August 4, 1993

Dr. William Travers
Deputy Associate Director for
Advanced Reactors and License Renewal
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, DC 20555

Dear Dr. Travers:

Enclosed please find the final PWR Reactor Vessel Internals and Low Voltage Environmentally Qualified Cable Industry Reports (IRs). Both reports have been revised pursuant to the Agreements-In-Principle (AIP) as reflected in the Comment Response Document (CRD) which are also enclosed.

The submittal of these two documents brings the total number of IRs transmitted to the NRC to ten.

In light of the June 28, 1993, Staff Requirements Memorandum (SRM) on license renewal, these revised IRs are submitted for the staff's use in incorporating the IR AIPs into the Standard Review Plan (SRP) for license renewal. We look forward to discussing the details of this process with the staff.

If you have any questions on the enclosed documents please contact Doug Walters.

Sincerely,

Raymond N. Ng
Manager, Technical Division

RNN/DJW/acm
Enclosures

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PRESSURIZED WATER REACTOR
REACTOR PRESSURE VESSEL INTERNALS
LICENSE RENEWAL INDUSTRY REPORT

NUMARC Report Number 90-05

Revision 1

December, 1992

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TABLE OF CONTENTS

Page

1.0 SUMMARY

1.1 Purpose 1-1
1.2 Scope 1-1
1.3 Methodology 1-2
1.4 Specific Conclusions for License Renewal 1-3
1.4.1 Nonsignificant Degradation Mechanisms 1-6
1.4.2 Nonsignificant Component/Degradation Mechanism Combinations . 1-7
1.4.3 Potentially Significant Component/Degradation Mechanism Combinations Managed by Effective Programs 1-10
1.4.4 Significant Component/Degradation Mechanism Combinations Requiring Plant-Specific Management 1-14

2.0 INTRODUCTION

2.1 Background 2-1
2.2 Purpose 2-1
2.3 Scope 2-1
2.4 Age-related Degradation Mechanisms 2-4
2.5 Evaluation Approach 2-4
2.6 License Renewal Applicant Responsibilities 2-5

3.0 INTERNALS COMPONENTS EVALUATION BASIS

3.1 Internals Components Design Features 3-1
3.1.1 Design Evolution 3-1
3.1.2 Internals Component Descriptions. 3-2
3.1.2.1 Westinghouse Internals Component Descriptions 3-11
3.1.2.2 Combustion Engineering Internals Component Descriptions. 3-15
3.1.2.3 B&W Internals Component Descriptions. 3-19
3.2 Internals Components Code, Standard and Regulation Bases. 3-23
3.2.1 General Design Background 3-23
3.2.2 Internals Load Categorization 3-23
3.2.3 System Hot Functional Test 3-24
3.2.4 Inservice Inspection (ISI) 3-25
3.2.5 Regulations and Regulatory Guidance Relevant to Internals Components 3-26
3.3 Operation and Maintenance History 3-27
3.3.1 Upper Internals Assembly 3-27
3.3.2 Core Support Assembly 3-28
3.3.3 Lower Internals Assembly 3-28
3.4 Section 3 References. 3-30

TABLE OF CONTENTS
(continued)

Page

4.0 AGE-RELATED DEGRADATION MECHANISM ASSESSMENT

4.1 Irradiation Embrittlment. 4-3
4.2 Stress Corrosion Cracking 4-6
4.3 Irradiation-assisted Stress Corrosion Cracking 4-11
4.4 Erosion and Erosion / Corrosion 4-14
4.5 Creep. 4-15
4.6 Stress Relaxation 4-16
4.7 Wear 4-19
4.8 Thermal Aging 4-21
4.9 Corrosion. 4-24
4.10 Fatigue 4-26
4.11 Degradation Mechanism Assessment Summary 4-32
4.12 Section 4 References. 4-36

5.0 EFFECTIVE PROGRAMS FOR POTENTIALLY SIGNIFICANT AGE-RELATED DEGRADATION

5.1 Irradiation Embrittlement 5-4
5.2 Stress Corrosion Cracking 5-5
5.3 Irradiation-assisted Stress Corrosion Cracking 5-6
5.4 Stress Relaxation 5-7
5.5 Wear 5-9
5.6 Thermal Aging 5-10
5.7 Fatigue. 5-12
5.7.1 Detailed Fatigue Analysis. 5-12
5.7.2 Fatigue Re-Evaluation 5-13
5.7.3 Fatigue Conclusions 5-14
5.8 Effective Program Assessment Summary. 5-15
5.9 Section 5 References 5-19

6.0 MANAGEMENT OPTIONS FOR POTENTIALLY SIGNIFICANT AGE-RELATED DEGRADATION

6.1 Assessment of Significance. 6-1
6.2 Aging Management Options 6-2
6.2.1 Mechanical Redundancy 6-2
6.2.2 Evaluation of Actual / Postulated Flaws. 6-3
6.2.3 Ultrasonic Testing 6-4
6.3 Section 6 References. 6-6

List of Tables

Table		Page
1.1	Component/Mechanism Disposition Summary	1-5
2.1	PWR RPV Internals Industry Report Scope	2-3
3.1	Chronology of U. S. Commercial Pressurized Water Reactors	3-5
3.2	Vendor Specific Component Designations	3-8
3.3	PWR RPV Internals Materials of Construction	3-9
4.1	Irradiation Embrittlement Summary	4-5
4.2	Stress Corrosion Cracking Summary	4-10
4.3	Irradiation-assisted Stress Corrosion Cracking Summary	4-13
4.4	Stress Relaxation Summary	4-18
4.5	Wear Summary	4-20
4.6	Thermal Aging Summary	4-23
4.7	Fatigue Summary	4-31
4.8	Degradation Mechanism Summary Identifying Components Carried Forward to Section 5	4-35
5.1	PWR Reactor Internals Components Requiring Section 5 Evaluation	5-3
5.2	Component / Mechanism Disposition Summary	5-18

List of Figures

Figure		Page
3.1	PWR RPV Internals Structural Assembly Groupings	3-7
3.2	Westinghouse RPV Internals System	3-10
3.3	Combustion Engineering RPV Internals System	3-14
3.4	Babcock & Wilcox RPV Internals System	3-18
4.1	The Effects of Oxygen and Chloride on the SCC of Austenitic Stainless Steels in High Temperature Water	4-8

ACRONYMS AND ABBREVIATIONS

ALARA	As Low As Reasonably Achievable
ANS	American Nuclear Society
ANSI	American National Standards Institute
ASME	American Society of Mechanical Engineers
ASTM	American Society for Testing and Materials
BMI	Bottom Mounted Instrumentation
B-N-3	Examination Category for Removable Core Support Structures in ASME Section XI, Subsection IWB
B&W	Babcock & Wilcox, B&W Nuclear Services Company
BWR	Boiling Water Reactor
CASS	Cast Austenitic Stainless Steel
CE	Combustion Engineering, ABB-Combustion Engineering
CEA	Control Element Assembly
CF 3	ASME/ASTM Material Grade Designation
CF 8	ASME/ASTM Material Grade Designation
CLB	Current Licensing Basis
CRA	Control Rod Assembly
CRDM	Control Rod Drive Mechanism
CS	Core Support Structure
CVTR	Carolina Virginia Tube Reactor
DOE	Department of Energy
ECM	Electro-Chemical Machining
ECT	Eddy-Current Testing
EPRI	Electric Power Research Institute
FIV	Flow Induced Vibration
GDC	General Design Criteria
HFIR	High Flux Isotope Reactor
IASCC	Irradiation-Assisted Stress Corrosion Cracking
ICI	In-Core-Instrumentation
IGSCC	Intergranular Stress Corrosion Cracking
IR	Industry Report
IS	Internals Structure
ISI	Inservice Inspection
IWB	Designation for the ASME Section XI Subsection containing requirements for Class 1 Components of Light-Water Cooled Power Plants
LCP	Lower Core Plate
LOCA	Loss of Coolant Accident
LSP	Lower Support Plate
MIMS	Metal Impact Monitoring System
Mo	Molybdenum
NG	Subsection of ASME Code, Section III Dealing With Core Support Structure
NRC	Nuclear Regulatory Commission

ACRONYMS AND ABBREVIATIONS
(continued)

NSSS	Nuclear Steam Supply System
NUMARC	Nuclear Management and Resources Council
ppb	parts per billion
PLEX	Plant Life Extension
PWR	Pressurized Water Reactor
R	Ratio of Minimum and Maximum Stress Intensities During Local Cycling
RC	Reactor Coolant
RCC	Rod Control Cluster
RCCA	Rod Control Cluster Assembly
RCPB	Reactor Coolant Pressure Boundary (per 10 CFR 50.2)
RCS	Reactor Coolant System
RMS	Root Mean Square
RPV	Reactor Pressure Vessel
RTD	Resistance Temperature Detector
SCC	Stress Corrosion Cracking
SS	Stainless Steel
T/C	Thermocouple
UCP	Upper Core Plate
USP	Upper Support Plate
UT	Ultrasonic Testing
VT-3	Designation for specific visual examination requirements under ASME Section XI
W, W	Westinghouse

1.0 SUMMARY

1.1 Purpose

The U.S. nuclear power industry, through coordination by the Nuclear Management and Resources Council (NUMARC), and sponsorship by the U.S. Department of Energy (DOE) and the Electric Power Research Institute (EPRI), has evaluated age-related degradation effects for a number of major plant systems, structures and components, in the license renewal technical Industry Reports (IRs). License renewal applicants may choose to reference these IRs in support of their plant-specific license renewal applications, as an equivalent to the integrated plant assessment provisions of the license renewal rule (10 CFR Part 54).

1.2 Scope

This IR provides the technical basis for license renewal for U.S. pressurized water reactor (PWR) reactor pressure vessel (RPV) internals components. The scope of the PWR RPV internals evaluated in this report includes those components that can be evaluated on a common functional basis among the three U.S. domestic PWR suppliers -- B&W Nuclear Service Company, ABB Combustion Engineering, or Westinghouse Electric. Such components have been identified as being important to PWR plant license renewal by pilot plant life extension (PLEX) studies carried out by Virginia Power, Baltimore Gas & Electric Company, Duke Power Company, and Yankee Atomic Electric Company, in cooperation with the Electric Power Research Institute (EPRI) and the U.S. Department of Energy (DOE). These pilot plant evaluations included a safety significance assessment of reactor vessel internals components and economic considerations associated with replacement and refurbishment, in addition to aging degradation technical issues. For the purposes of this report, PWR RPV internals do not include components which are attached to, or are an integral part of the reactor vessel or head.

For the evaluations in this report, the internals components are divided into three structural assembly groupings:

(a) the Upper Internals Assembly

(b) the Core Support Assembly

(c) the Lower Internals Assembly

The Upper Internals Assembly consists of all the internals components above the core which are removed at each refueling. For the purposes of this report, the remaining internals components have been arbitrarily divided into the Core Support Assembly and the Lower Internals Assembly, and none of these components are generally removed at refueling. The Lower Internals Assembly consists of all components below the core; the Core Support Assembly consists of the remaining components. The Upper Internals Assembly, Core Support Assembly and Lower Internals Assembly groupings are not necessarily the

designations used by the three vendors; they are used here solely to provide a common framework for discussion.

Table 1.1 identifies the vendor specific designations of internals components covered by this report and contained within the upper internals, core support, and lower internals assemblies. In Table 1.1, components which appear in the same row have the same general function in Westinghouse, Combustion Engineering, and Babcock and Wilcox internals designs. The absence of an entry means that a specific vendor design does not have such a component. For those components which have subcomponents which may be affected by specific age-related degradation mechanisms, these subcomponents are indented under the major component heading. Similarity in function does not extend to subcomponents which may appear on the same line. The specific scope of the report is discussed in further detail in Section 2.

1.3 Methodology

The license renewal technical evaluation consists of four parts: First, the component evaluation basis is established (Section 3), consisting of component descriptions, the general component design bases, and relevant component operation history. Second, the age-related degradation mechanisms that could affect these components are described and their potential significance to performance of component safety function(s) performance, as defined in 10 CFR Part 100, Appendix A, paragraph (a) of section VI, is evaluated (Section 4). Third, for combinations of age-related degradation mechanisms and components that are determined to have a potentially significant effect on component safety function, the capability of effective programs to manage the potentially significant effects of age-related degradation is examined (Section 5). Fourth, for cases where generic effective programs cannot be shown to be capable of managing the effects of age-related degradation, aging management options for plant-specific programs are described (Section 6).

The age-related degradation mechanisms to which PWR RPV internals components could be subjected are addressed in this report and are listed below:

1. Irradiation Embrittlement.
2. Stress Corrosion Cracking (SCC).
3. Irradiation-assisted Stress Corrosion Cracking (IASCC).
4. Erosion and Erosion/Corrosion.
5. Creep.
6. Stress Relaxation.
7. Wear.
8. Thermal Aging
9. Corrosion.
10. Fatigue.

These mechanisms were identified from a review of vendor-specific evaluations, nuclear power plant operation experience, relevant laboratory data, and related experience in other industries.

No consideration has been made that these age-related degradation mechanisms are or are not unique to the license renewal term. Such an evaluation is the responsibility of the license renewal applicant.

An age-related degradation mechanism is considered significant if, when allowed to continue without any additional prevention or mitigation measures, it cannot be shown that the component would maintain its safety function during the license renewal period. The potential for significant age-related degradation of specific components evaluated in this IR is dependent on their design features (Section 3.1), their design bases (Sections 3.2), their operation and maintenance histories (Section 3.3), and the extent to which they are susceptible to the age-related degradation mechanisms (Section 4). If it can be shown that the PWR reactor vessel internals components are either not susceptible to the degradation mechanisms under consideration, or is susceptible to such a small degree that the specific component safety function is not adversely affected throughout the license renewal term, then the component/degradation mechanism combination is not significant (Section 4). Otherwise, the component/degradation mechanism combination is potentially significant. If a potentially significant component/degradation mechanism combination is adequately addressed by effective programs, as justified in the report, then the issue is considered to be resolved on the basis that the degradation is managed acceptably (Section 5). Combinations of mechanisms and components for which generic effective program elements cannot be shown to manage potentially significant age-related degradation require plant-specific evaluation. Recommendations for plant specific aging management options for such issues are then provided (Section 6), if required.

A license renewal applicant intending to take credit for a particular conclusion of this report is responsible for the review/evaluation of plant-specific features, including appropriate current licensing basis (CLB) documents/information, to assure that the assumptions and criteria used as the basis for the determination of insignificance in Section 4 of this report are applicable to the component under consideration. Similarly, the license renewal applicant is responsible for the review/evaluation of their plant-specific features, including appropriate CLB documents/information, in order to assure that the program elements used to manage the effects of potentially significant degradation in Section 5 of this report, or their justified equivalent, are committed for use at their plant. Plant-specific evaluations are required for any age-related degradation/component issues for which the license renewal applicant is unable to demonstrate that the generic stipulated program elements are committed for use, or the basis for the conclusions are applicable, at their plant.

1.4 Specific Conclusions for License Renewal

A summary of the specific conclusions for license renewal is provided in the following subsections. These conclusions are grouped as follows, based on the method of resolution and the corresponding IR section where they are resolved:

Types of Conclusions

- Age-related degradation mechanisms that are considered to be non-significant for any component (Summarized in 1.4.1, Details in Section 4).
- Age-related degradation mechanisms that are considered to be non-significant for specific components (Summarized in 1.4.2, Details in Section 4).
- Potentially significant age-related degradation mechanisms for specific components that are managed by effective programs (Summarized in 1.4.3, Details in Section 5).
- Significant age-related degradation mechanisms for specific components that are subject to plant-specific management (Summarized in 1.4.4, Details in Section 6).

This PWR RPV Internals Industry Report evaluated ten age-related degradation mechanisms potentially applicable to thirteen components (not counting subcomponents) for each of the three major U.S. PWR NSSS vendors.

In general, the age-related degradation mechanisms affect different components or groups of components differently. This results in an evaluation of the effects of 10 mechanisms on 13 components for each of the 3 NSSS vendors or a component-degradation mechanism matrix of 10x13x3 elements as shown in Table 1.1. The evaluations in Sections 4, 5, and 6 address each degradation mechanism for each of the reactor vessel internals components as well as key subcomponents known to be susceptible to specific age-related degradation. The following summary sections are also presented in this manner.

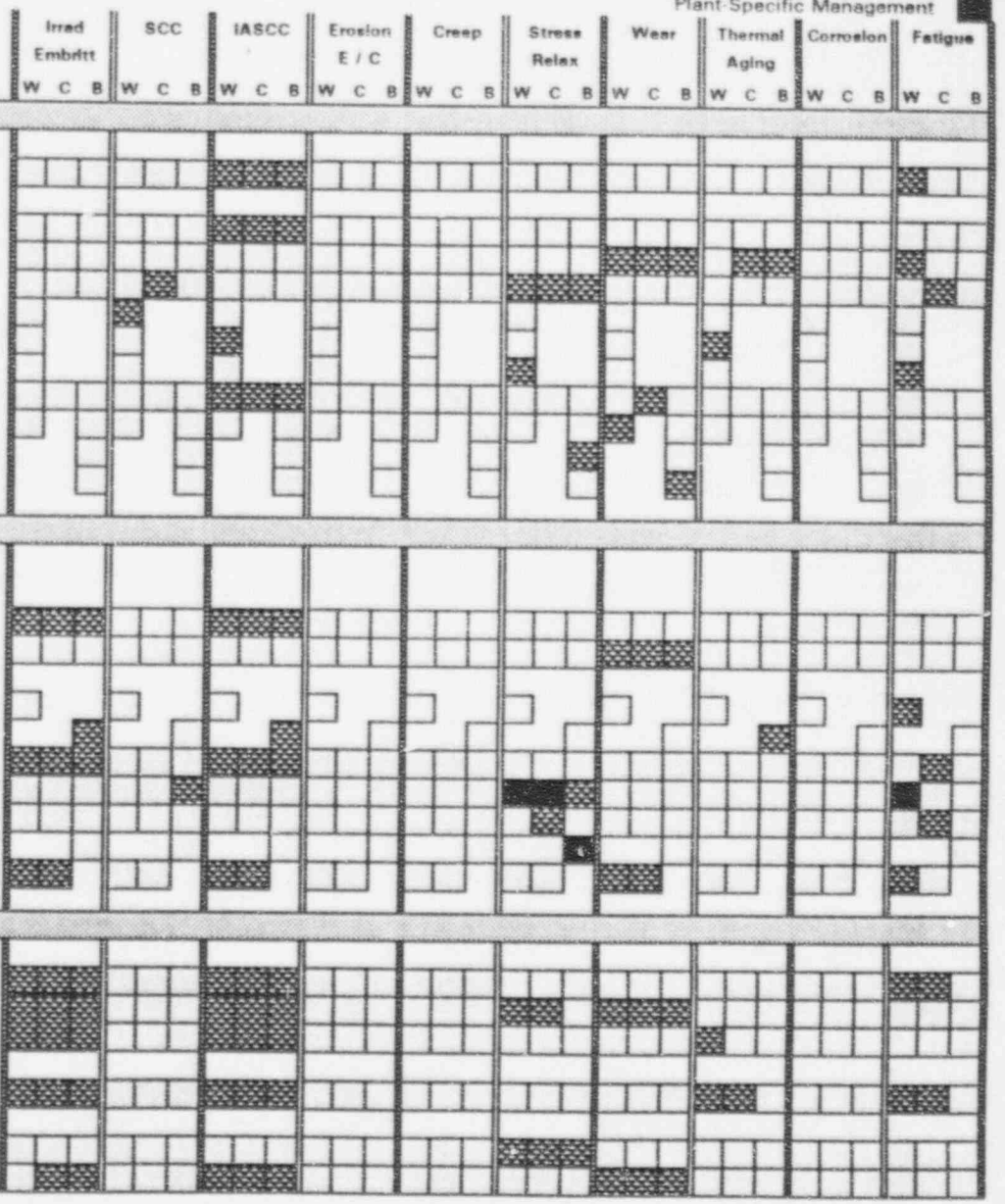
Table 1.1

COMPONENT / MECHANISM DISPOSITION SUMMARY

for each STRUCTURAL ASSEMBLY GROUPING

Key: No Significant Degradation; Effective Program; Plant-Specific Management

WESTINGHOUSE (W)	COMBUSTION ENGINEERING (C)	BABCOCK & WILCOX (B)
UPPER INTERNALS ASSEMBLY		
Upper Internals Assembly	**Upper Guide Structure Assembly**	**Plenum Assembly**
1. Upper Support Plate	1. Upper Guide Structure Support Plate	1. Plenum Cover and Plenum Cylinder
2. RCCA Guide Tube Assemblies	2. CEA Shroud Assemblies	2. CRA Guide Tube Assemblies
RCCA Guide Tubes	CEA Shrouds	CRA Guide Tubes
RCCA Guide Tube Bolts	CEA Shroud Bolts	CRA Guide Tube Bolts
RCCA Guide Tube Support Pins		
3. Upper Support Columns	3. ----------	3. ----------
Upper Support Column Bolts		
4. Upper Core Plate	4. Fuel Alignment Plate	4. Upper Grid Assembly
Fuel Pins		Upper Grid Rib Section
		Upper Grid Assembly Bolts
		Fuel Guide Pads
CORE SUPPORT ASSEMBLY		
Core Barrel and Baffle/Former Assembly	**Core Support Barrel and Core Shroud Assembly**	**Core Support Assembly**
1. Core Barrel	1. Core Support Barrel	1. Core Support Shield
Upper Core Barrel Flange	Core Support Barrel Upper Flange	Core Support Shield Flange
Core Barrel Nozzles		
2. ----------	2. ----------	2. Vent Valve Assemblies
3. Baffle-Former Assembly	3. Core Shroud Assembly	3. Core Barrel Assembly
Baffle / Former Assembly Bolts	Core Shroud Assembly Bolts	Core Barrel Bolts
Baffle / Former Assembly Baffles	Core Shroud Tie Rods	Baffle / Former Plates
		Baffle / Former Bolts
4. Upper Core Plate Alignment Pins	4. Fuel Alignment Plate Guide Lugs	4. ----------
LOWER INTERNALS ASSEMBLY		
Lower Internals Assembly	**Core Support Assembly**	**Lower Grid Assembly**
1. Lower Core Plate	1. Core Support Plate	1. Lower Grid Top Rib Section
2. Fuel Pins	2. Fuel Alignment Pins	2. Fuel Guide Pads
3. Lower Support Plate	3. Lower Support Structure Beam Assemblies	3. Lower Grid Bottom Rib Weldment
4. Lower Support Columns	4. Core Support Columns	4. Lower Grid Assembly Support Posts
Lower Support Column Bolts	Core Support Column Bolts	Lower Grid Assembly Bolts
5. Radial Keys and Clevis Inserts	5. Core Support Barrel Snubber Assemblies	5. Lower Grid Cylinder Guide Blocks



1.4.1 Nonsignificant Degradation Mechanisms

A first set of issues is considered to be generically resolved, on the basis that the aging mechanism being evaluated does not cause significant degradation for any PWR reactor vessel internals components covered by this report. License renewal applicants intending to reference these generic conclusions are responsible for a review/evaluation of plant specific features, including appropriate CLB documents/information, in order to assure that there are no deviations from the assumptions and criteria used in the report.

- **Erosion and Erosion/Corrosion** (Section 4.4.2) All of the PWR RPV internals components being considered here are constructed of austenitic stainless steel or nickel-base alloys that are considered to be resistant to erosion and erosion/corrosion in a PWR environment. The relatively low fluid flow velocities in the PWR environment do not permit erosion to become a significant age-related degradation issue. The pH levels in the bulk coolant minimize corrosion and thus erosion/corrosion. The operating pressures of a PWR (approximately 2250 psia) preclude cavitation erosion, and the purity and particulate control of the reactor coolant eliminate particulate erosion as a significant concern [4-18]. Therefore, erosion and erosion/corrosion are nonsignificant age-related degradation mechanisms for any of the PWR reactor internals components considered in this report.

- **Creep** (Section 4.5.2) The maximum temperature experienced by PWR RPV internals during normal and upset conditions is approximately 650°F, except for certain localized areas where temperatures can be as high as 800°F for short durations. Even with localized temperature excursions to 800°F, these temperatures are well below the temperatures at which creep is a concern for any of the austenitic stainless steel and nickel-base alloy PWR RPV internals components. Therefore, creep is a nonsignificant age-related degradation mechanism for any PWR RPV internals component.

- **Corrosion** Section (4.9.2) Corrosion is an electrochemical reaction between a metal or alloy and its environment and is characterized by a deterioration of the materials. Two forms of corrosion are typically defined: (1) general corrosion, and (2) pitting or crevice corrosion. The materials used in the construction of the reactor internals components are austenitic stainless steel and nickel-base alloys. These alloys are not susceptible to general corrosion in the benign PWR primary coolant because they passivate to form protective layers which mitigate the potential for corrosion degradation.

  This resistance extends to crevice regions, such as bolt threads, where an aggressive environment has the potential to cause localized corrosion, even for these film-forming materials. In many crevice tests conducted at elevated temperature with different materials, it was found that the extent of crevice corrosion generally decreased with the increasing corrosion resistance of the material tested. Straight chromium stainless steels and nickel-alloys were subject to slight pitting in crevice

areas, whereas austenitic stainless steels were not. The majority of the internals components containing crevices, such as baffle and former assemblies and fasteners, are fabricated from austenitic stainless steel alloys and hence not anticipated to be susceptible to pitting.

Sufficient data exists to indicate that the hydrogen overpressure environment present in the reactor vessel plays an important role in the control of crevice corrosion in PWRs by minimizing the adverse effects of oxygen through recombination. In addition to the control of radiolytic dissolved oxygen in the regions of the core by excess dissolved hydrogen, stagnant (crevice) regions are protected by: (1) diffusion of hydrogen from the bulk coolant, (2) enhanced recombination rates from the effects of the crevice on the radiation quality, and (3) rapid reaction of the oxygen with the materials of construction leaving an excess of hydrogen in the stagnant areas. Therefore, it is concluded that adequate protection exists to mitigate the effects of crevice corrosion in PWR RPV internals components. No further evaluation is required with respect to corrosion for the PWR RPV internals components fabricated from austenitic stainless steel or nickel-base alloys.

1.4.2 Nonsignificant Component/Degradation Mechanism Combinations

A second set of issues is considered to be generically resolved, since the aging mechanism being evaluated does not cause significant degradation to specific PWR reactor vessel internals components. License renewal applicants intending to reference these generic conclusions are responsible for a review/evaluation of plant-specific features, including appropriate CLB documents/information, in order to assure that there are no deviations from the assumptions and criteria used in the report.

- **Irradiation Embrittlement** (Section 4.1.2) Irradiation embrittlement can occur in PWR RPV internals components fabricated from austenitic stainless steel and nickel-base alloys with expected neutron fluences in excess of 1×10^{20} n/cm^2 ($E > 1$ MeV). The internals components most susceptible to irradiation embrittlement are those which are nearest to the reactor core. These components will experience significant neutron irradiation exposure, while remotely located components will receive relatively little. For those PWR RPV internals components experiencing neutron fluence less than the threshold value of 1×10^{20} n/cm^2 ($E > 1$ MeV), the changes in mechanical properties due to neutron exposure are nonsignificant and no further evaluation of these components is required for the purposes of license renewal. The PWR RPV internals components for which irradiation embrittlement is potentially significant are given in Table 1.1 and evaluated further in Section 5.
- **Stress Corrosion Cracking** (Section 4.2.2) Initiation and propagation of SCC requires three factors to be present: (1) a susceptible material, (2) a corrosive environment and (3) the presence of tensile stresses. For reactor internals components fabricated from austenitic stainless steel, provided that the components are subjected to stress levels within the material design specification and the PWR operating chemistry can be shown to limit the oxygen to < 5 ppb and the halogens

to < 150 ppb, then SCC is a nonsignificant age-related degradation mechanism and no further evaluation of SCC for the purposes of license renewal is required. Otherwise, further evaluation is required and is provided in Section 5.

Stress corrosion cracking (SCC) is not a concern for PWR RPV internals components fabricated from austenitic stainless steel materials with a ferrite content of 5% or more. This ferrite content reduced the susceptibility to SCC. No further evaluation of SCC for the purposes of license renewal is required for PWR RPV internals components fabricated of austenitic stainless steel that have a ferrite content of 5% or more.

For reactor internals components fabricated from nickel-base alloys, SCC is a potentially significant age-related degradation mechanism and further evaluation of these components is provided in Section 5.

Two types of reactor internals components which have experienced SCC degradation are austenitic stainless steel Alloy A-286 fasteners in the B&W plants and nickel-base Alloy X-750 support pins in Westinghouse plants. Fasteners within the reactor internals identified in Table 1.1 potentially fall within the two materials groupings described above where all three factors required for the initiation and propagation of SCC are present. These fasteners are evaluated further in Section 5.

- **Irradiation-assisted Stress Corrosion Cracking** (Section 4.3.2) Irradiation-assisted stress corrosion cracking (IASCC) may be a potentially significant age-related degradation mechanism for PWR RPV internals components. IASCC is not a significant age-related degradation mechanism provided the primary coolant is controlled to PWR operating limits, the fluence level is less than 5×10^{20} n/cm^2 for materials with low (< 10 ksi) stress levels (1×10^{20} n/cm^2 for moderate to high (> 10 ksi) stress levels) and the operating stresses are within the design envelope. For PWR RPV internals components that are not enveloped by the above conditions, IASCC is a potentially significant age-related degradation mechanism. Neutron flux maps or measurements, if available, may be used to show the fluence level is below the threshold levels (5×10^{20} or 1×10^{20} n/cm^2). However, for the purposes of this report, it will be assumed that fluence levels exceed these threshold values and all PWR RPV internals components will be considered potentially susceptible to IASCC. Further evaluation is required and is provided in Section 5.

- **Stress Relaxation** (Section 4.6.2) Stress relaxation has significance only to components with substantial preloads, such as torqued bolts, because the maintenance of an adequate preload is important to their functionality. A loss of preload in such components could result in higher cyclic and transient loads and an increase in fatigue susceptibility (Section 4.10). In some reactor internals components, stress relaxation is accounted for in the design by preload margins that are incorporated to compensate for the effects of thermal transients. Neutron irradiation can, over an extended period of time and depending on the fluence, lead to stress relaxation of preloaded components. Therefore, provided that the internals components do not depend on preload for functionality (e.g., bolting), stress

relaxation is a nonsignificant age-related degradation mechanism affecting the license renewal of PWR reactor internals components. The PWR RPV internals components for which stress relaxation is potentially significant are identified in Table 1.1. Otherwise, stress relaxation is potentially significant and requires further evaluation as provided in Section 5.

- **Wear** (Section 4.7.2) Wear has been observed at the interfaces of components whose relative motion is not completely restrained. In the reactor internals components, wear between control rods and guide tubes results from the axial sliding which occurs during insertions and withdrawals, and also from the transverse motions caused by flow-induced vibration. Other components considered to be potentially susceptible to wear degradation are those which constitute the interfaces between structural components. The predominant excitation mechanism for wear of these components is flow-induced vibration, although thermal effects can also contribute by increasing interface loads and by directly producing relative thermal displacements between interfacing components. Wear is a potentially significant age-related degradation mechanism for the internals components identified in Table 1.1. Further evaluation of wear with respect to these components is provided in Section 5. For all other PWR RPV internals components covered by this report, wear is nonsignificant.

- **Thermal Aging** (Section 4.8.2) Thermal aging embrittlement is a potentially significant age-related degradation mechanism for the cast austenitic stainless steel (CASS) PWR RPV internals components only for the combination of metallurgical conditions and manufacturing process which exceed the following threshold criteria.

  Thermal aging embrittlement is potentially significant for :

  (1) Centrifugally-cast component parts, with a delta ferrite content above 20%.

  (2) Statically-cast component parts, with Mo content meeting CF3 and CF8 limits and with a delta ferrite content above 20%.

  (3) Statically-cast component parts, with Mo content exceeding CF3 and CF8 limits with a delta ferrite content above 14%.

  Thermal aging embrittlement is also a potentially significant age-related degradation mechanism for the martensitic stainless steel subcomponents in the B&W vent valve assemblies. Evaluation of these components is provided in Section 5. For all other PWR RPV internals components covered by this report, thermal aging is nonsignificant. No further evaluation is required with respect to thermal aging for the PWR RPV internals components not explicitly defined above.

- **Fatigue** (Section 4.10.2) For license renewal evaluation purposes, fatigue will be considered to be a nonsignificant age-related degradation mechanism for a component when two criteria are met: (1) The component must have no history of fatigue damage and (2) the fatigue usage factor must be below any threshold of concern. Any component that does not meet these two criteria will be considered a

potentially fatigue-critical component which requires further evaluation for license renewal purposes. For the definition of fatigue-critical, the conservative fatigue usage factor threshold of concern is 0.4.

For those plants with PWR RPV internals components with an explicit Section III fatigue design basis, or its equivalent, fatigue-critical internals components can be identified, for license renewal purposes, from plant design stress reports or hot functional test data reports. For earlier plants whose internals components did not have an explicit fatigue design basis, the principles of similitude apply. Similitude demonstration is based upon a number of factors, foremost of which are similarity of geometry and similarity of component operating history. For example, Babcock & Wilcox [4-3] used the explicit fatigue analysis results for their 205 FA plant RPV internals and their similarity to geometries and operating conditions for 177 FA plant RPV internals to assess fatigue damage in an early plant. The fatigue-critical components were identified and these components were then reanalyzed to include a realistic (RMS) evaluation of cyclic amplitudes, with the new fatigue usage factors found to be considerably lower than the originally-calculated fatigue usage factors.

Fatigue-critical internals components have also been identified for Westinghouse and Combustion Engineering plants, using design stress report and hot functional test data for later plants.

Therefore, with the exception of those PWR RPV internals components identified in Table 1.1 as fatigue-critical -- based upon a review of calculated fatigue usage factors for internals components designed to ASME Section III, Subsection NG; a review of hot functional test data; and a comparison of geometric and operating similarities -- fatigue is not a significant age-related degradation mechanism for PWR internals components. Fatigue-critical PWR internals components require further evaluation as provided in Section 5.

1.4.3 Potentially Significant Component/Degradation Mechanism Combinations Managed by Effective Programs

A third set of conclusions covers issues that are considered to be generically resolved for specific PWR reactor vessel internals components, because any potentially significant age-related degradation can be shown to be within established limits by acceptable maintenance, inspection, test, and analytical evaluation procedures. A license renewal applicant intending to take credit for an effective program is responsible for the review/evaluation of appropriate plant-specific features, including appropriate CLB documents/information, in order to assure that the program elements required to manage the effects of potentially significant age-related degradation, or their justified equivalent, continue being used at their plant.

- **Irradiation Embrittlement** (Section 5.1) Irradiation embrittlement is a potentially significant age-related degradation mechanism for PWR RPV internals components fabricated from austenitic stainless steel and nickel-base alloys which experience neutron fluences in excess of 1×10^{20} n/cm^2 (E > 1 MeV). The

internals components most susceptible to irradiation embrittlement are those located nearest to the reactor core. These components are evaluated herein as a part of the core support assembly and the lower internals assembly and have been identified in Table 1.1. For irradiation embrittlement, the relevant conditions related to cracked or fractured parts are of most interest, since the anticipated manifestation of neutron irradiation embrittlement damage is a lower resistance to crack initiation. As a result of ASME Code Section XI visual (VT-3) examination standards with respect to demonstrated character recognition height and minimum distance from the examiner's eye to the object surface, adequate detectability of surface flaws of significant length is assured. Therefore, VT-3 inservice examination of core support assembly and lower internals assembly PWR RPV internals components, in accordance with the requirements for examination category B-N-3 of Subsection IWB, Section XI, is able to manage the effects of irradiation embrittlement, provided that such components are accessible, or can be rendered accessible by the removal of the core and/or other internals, for the examination.

- **Stress Corrosion Cracking** (Section 5.2) Stress corrosion cracking (SCC) is a potentially significant age-related degradation mechanism for PWR RPV internals components which are fabricated from Alloy A-286 or Alloy X-750 materials. B&W Alloy A-286 core barrel bolts, CE Alloy A-286 CEA shroud bolts and Westinghouse Alloy X-750 RCCA guide tube support pins were the components identified in the report which are potentially affected by SCC. For the case of Alloy A-286 core barrel bolts at Babcock & Wilcox plants, a limited number of Alloy A-286 core barrel bolt failures have been observed that may be attributable to SCC. In all cases where such failures have occurred, the affected and potentially susceptible bolts have been replaced by bolts that have been designed to maintain stress levels below the SCC threshold for initiation or that have been fabricated from materials less susceptible to SCC, or both. In addition, Babcock & Wilcox has recommended periodic inspection programs to plant owners in order to monitor potential age-related degradation associated with SCC of core barrel bolts.

  CEA shroud bolts for Combustion Engineering plant are also fabricated from Alloy A-286 material. In this case, since the upper internals are removed at each refueling outage, visual inspection in accordance with the procedures of ASME Section XI, Subsection IWB provides assurance that the effects of potential SCC of the CEA shroud bolts are detected and the age-related degradation is effectively managed.

  RCCA guide tube support pins in Westinghouse plants are fabricated from Alloy X-750. Some SCC and eventual fracture of these pins have been observed in service and attributed to the combination of high stress and a susceptible material condition. In the majority of the affected plants, these Alloy X-750 support pins have been replaced by pins designed to reduce the applied stresses below the SCC threshold and fabricated from a material with a heat treatment to reduce susceptibility. All of the affected utilities have developed and put procedures in place to detect and control any incipient SCC degradation of these components.

Therefore, visual (VT-3) inservice examination of Alloy A-286 core barrel bolts in B&W plants, Alloy A-286 CEA shroud bolts in Combustion Engineering plants, and Alloy X-750 RCCA guide tube support pins, in accordance with the requirements for examination category B-N-3 of Subsection IWB, Section XI, is able to manage the effects of stress corrosion cracking, provided that such components are accessible, or can be rendered accessible by the removal of the core and/or other internals, for the examination.

- **Irradiation-assisted Stress Corrosion Cracking** (Section 5.3)
  Irradiation-assisted stress corrosion cracking (IASCC) was evaluated in Section 4.3 to be a potentially significant age-related degradation mechanism for PWR RPV internals components. It was assumed that fluence levels experienced by all PWR RPV internals components under review here exceeded the threshold values established for nonsignificance. In Section 5.3, inservice inspection is shown to be an effective program that can be used to manage IASCC. ASME Section XI, Subsection IWB, Examination Category B-N-3 provides requirements for the visual (VT-3) examination of "removable core support structures" of PWRs. These requirements refer to the relevant conditions defined in IWB-3520.2, which include "loose, missing, cracked, or fractured parts, bolting, or fasteners." If any relevant condition is detected, IWB-3142 provides options for correcting the relevant condition, such as: (i) acceptance by supplemental surface and/or volumetric examination, in order to characterize the indication more accurately, (ii) acceptance by analytical evaluation, which may involve more frequent examination of the item; or (iii) acceptance by replacement of the item. Therefore, IASCC of all accessible PWR RPV internals components is effectively managed by inservice inspection under the requirements of ASME Section XI, Subsection IWB, Examination Category B-N-3. For those components that are not accessible, plant-specific actions are described in Section 6.2.3.

- **Stress Relaxation** (Section 5.4) Stress relaxation is a potentially significant age-related degradation mechanism for PWR RPV internals components that require adequate preload to perform their intended safety function. The components that fall into this category have been identified in Table 1.1 and are discussed further in Section 5.4. Since the manifestation of excessive stress relaxation is expected to be loose, cracked or missing bolts or fasteners, the ASME Section XI visual (VT-3) examination is adequate for the detection of significant stress relaxation and loss of preload. For those few cases where the relevant conditions of ASME Section XI, Subsection IWB, Examination Category B-N-3 may be unable to detect significant stress relaxation and loss of preload, because the assurance of functional capability cannot be guaranteed by detection of loose, missing, cracked, or fractured parts, bolting, or fasteners, alternatives for managing the effects of age-related degradation are described in Section 6. Baffle-former assembly bolts in Westinghouse and Babcock & Wilcox plants, and core shroud assembly bolts in Combustion Engineering plants, have been deemed to be in this category. With these exceptions, ASME Section XI visual (VT-3) inservice examination of PWR RPV internals components that require adequate preload in order to perform their

intended safety function, in accordance with the requirements for Examination Category B-N-3 of Subsection IWB, Section XI, is able to manage the effects of stress relaxation, provided that such components are accessible, or can be rendered accessible by the removal of the core and/or other internals, for the examination.

- **Wear** (Section 5.5) Wear is a potentially significant age-related degradation mechanism for PWR RPV internals components which may have relative motion with adjacent surfaces. Since the manifestation of excessive wear is expected to be wear of mating surfaces that may lead to loss of function, the ASME Section XI visual (VT-3) examination is adequate for the detection of significant wear. Wear between two surfaces subject to relative motion is readily detectable by visual inspection long before the effects of wear begin to compromise the structural integrity or function of the component. If it is determined that excessive wear has taken place, by visual inspection or code-defined alternate NDE means, subsequential wear evaluation can be performed to determine the need for refurbishment or repair of the worn areas. Therefore, visual (VT-3) inservice examination of PWR reactor internals components that may have relative motion with respect to adjacent surfaces, in accordance with the requirements for Examination Category B-N-3 of Subsection IWB, Section XI, is able to manage the effects of wear.

- **Thermal Aging** (Section 5.6) Thermal aging embrittlement is a potentially significant age-related degradation mechanism for the cast austenitic stainless steel (CASS) PWR RPV internals components only for the combination of metallurgical conditions and manufacturing process which exceed the following threshold criteria.

  Thermal aging embrittlement is potentially significant for :

  (1) Centrifugally-cast component parts, with a delta ferrite content above 20%.

  (2) Statically-cast component parts, with Mo content meeting CF3 and CF8 limits and with a delta ferrite content above 20%.

  (3) Statically-cast component parts, with Mo content exceeding CF3 and CF8 limits with a delta ferrite content above 14%.

  Thermal aging embrittlement is also a potentially significant age-related degradation mechanism for the martensitic stainless steel subcomponents in the B&W vent valve assemblies. Since the manifestation of excessive thermal aging is expected to be a reduced resistance to crack initiation, the visual (VT-3) inservice examination required by ASME Section XI, Subsection IWB Examination Category B-N-3 for the PWR RPV internals components is able to manage the effects of thermal aging on these components fabricated from cast austenitic stainless steel or martensitic stainless steel.

- **Fatigue** (Section 5.7) Fatigue is a potentially significant age-related degradation mechanism for PWR RPV internals components found to be fatigue-critical. The fatigue-critical components requiring further evaluation are listed in Table 1.1. For

these components, two evaluation methods are available: (1) a detailed fatigue evaluation/re-evaluation under the rules of ASME Section III, Subsection NG and (2) inservice examination using ASME Section XI, Subsection IWB, Examination Category B-N-3 procedures. From the investigations done during preparation of this report, the only fatigue-critical component requiring further plant-specific evaluation is the Westinghouse baffle / former bolts. Fatigue of other PWR RPV internals components judged to be fatigue-critical can be effectively managed using the two methods identified above and described more fully in Section 5.7.

1.4.4 Potentially Significant Component/Degradation Mechanism Combinations Requiring Plant-Specific Management

A final set of PWR reactor internals component / degradation mechanism combinations has been found to be potentially significant and no effective program has been identified to manage the degradation. Applicants for license renewal must perform a plant-specific evaluation for these component / degradation mechanism combinations. These combinations are (1) stress relaxation of one set of bolts in the Core Support Assembly component grouping (specifically, Westinghouse baffle/former assembly bolts, CE core shroud assembly bolts and B&W baffle/former assembly bolts) and (2) fatigue of Westinghouse baffle/former assembly bolts. Plant-specific options exist to manage both of these issues. These options include taking credit for redundancy in these multiple structure support systems, evaluation of actual/postulated flaws in these components and applying ultrasonic testing techniques to determine the presence of flaws in these bolts. More detailed discussion of these options is provided in Section 6.2

Additionally, for those individual cases where a utility determines that the effective program elements outlined in Section 5 are not fully implementable or where a utility may judge that the effective program elements are not fully effective for that utility's specific design, then alternatives to these program elements may also be available. A few of these alternatives are covered by the aging management options presented in Section 6.2.

2.0 INTRODUCTION

2.1 Background

This report provides a technical basis for an evaluation of the reactor pressure vessel (RPV) internals of a pressurized water reactor (PWR) that supports continued plant operation during the license renewal period. This technical basis is intended to support a plant owner's license renewal submittal. The RPV internals designs of the three major PWR NSSS vendors -- B&W Nuclear Service Company, ABB Combustion Engineering (CE), and Westinghouse -- are addressed herein.

In this section, the specific PWR RPV internals components within the scope of the report will be defined along with the age-related degradation mechanisms which can affect these components. The evaluation process of the report will then be outlined and the responsibilities of the license renewal applicant in the use of this report's findings will be delineated.

2.2 Purpose

This industry report (IR) provides a generic technical basis for license renewal of specific PWR RPV internals components. The generic evaluations have the objective of demonstrating the capability of systems, structures and components to continue to perform their intended safety functions throughout the license renewal term.

2.3 Scope

The scope of the PWR RPV internals evaluated in this report includes as practical those components that can be evaluated on a common functional basis among the three U.S. domestic PWR vendor designs. For the purposes of this report, PWR RPV internals do not include components which are attached to, or are an integral part of the reactor vessel or head.

For the evaluations in this report, the internals components are divided into three structural assembly groupings:

(a) the Upper Internals Assembly

(b) the Core Support Assembly

(c) the Lower Internals Assembly

The Upper Internals Assembly consists of all the internals components above the core which are removed at each refueling. For the purposes of this report, the remaining internals components have been arbitrarily divided into the Core Support Assembly and the Lower Internals Assembly, and none of these components are generally removed at refueling. The Lower Internals Assembly consists of all components below the core; the Core Support Assembly consists of the remaining components. The Upper Internals Assembly, Core

Support Assembly and Lower Internals Assembly groupings are not necessarily the designations used by the three vendors; they are used here solely to provide a common framework for discussion.

Table 2.1 identifies the vendor specific designations of internals components covered by this report and contained within the Upper Internals, Core Support, and Lower Internals Assemblies. In Table 2.1, components which appear in the same row have the same general function in Westinghouse, Combustion Engineering, and Babcock and Wilcox internals designs. The absence of an entry means that a specific vendor design does not have such a component. For those components which have subcomponents which may be affected by specific age-related degradation mechanisms, these subcomponents are indented under the major component heading. Similarity in function does not extend to subcomponents which may appear on the same line.

Table 2.1

PWR RPV INTERNALS
INDUSTRY REPORT SCOPE

Westinghouse	Combustion Engineering	Babcock & Wilcox
UPPER INTERNALS ASSEMBLY		
Upper Internals Assembly	**Upper Guide Structure Assembly**	**Plenum Assembly**
1. Upper Support Plate	1. Upper Guide Structure Support Plate	1. Plenum Cover and Plenum Cylinder
2. RCCA Guide Tube Assemblies RCCA Guide Tubes RCCA Guide Tube Bolts RCCA Guide Tube Support Pins	2. CEA Shroud Assemblies CEA Shrouds CEA Shroud Bolts	2. CRA Guide Tube Assemblies CRA Guide Tubes CRA Guide Tube Bolts
3. Upper Support Columns Upper Support Column Bolts	3. ----------	3. ----------
4. Upper Core Plate Fuel Pins	4. Fuel Alignment Plate	4. Upper Grid Assembly Upper Grid Rib Section Upper Grid Assembly Bolts Fuel Guide Pads
CORE SUPPORT ASSEMBLY		
Core Barrel and Baffle/Former Assembly	**Core Support Barrel and Core Shroud Assembly**	**Core Support Assembly**
1. Core Barrel Upper Core Barrel Flange Core Barrel Nozzles	1. Core Support Barrel Core Support Barrel Upper Flange	1. Core Support Shield Core Support Shield Flange
2. ----------	2. ----------	2. Vent Valve Assemblies
3. Baffle-Former Assembly Baffle / Former Assembly Bolts Baffle / Former Assembly Baffles	3. Core Shroud Assembly Core Shroud Assembly Bolts Core Shroud Tie Rods	3. Core Barrel Assembly Core Barrel Bolts Baffle / Former Plates Baffle / Former Bolts
4. Upper Core Plate Alignment Pins	4. Fuel Alignment Plate Guide Lugs	4. ----------
LOWER INTERNALS ASSEMBLY		
Lower Internals Assembly	**Core Support Assembly**	**Lower Grid Assembly**
1. Lower Core Plate	1. Core Support Plate	1. Lower Grid Top Rib Section
2. Fuel Pins	2. Fuel Alignment Pins	2. Fuel Guide Pads
3. Lower Support Plate	3. Lower Support Structure Beam Assemblies	3. Lower Grid Bottom Rib Weldment
4. Lower Support Columns Lower Support Column Bolts	4. Core Support Columns Core Support Column Bolts	4. Lower Grid Assembly Support Posts Lower Grid Assembly Bolts
5. Radial Keys and Clevis Inserts	5. Core Support Barrel Snubber Assemblies	5. Lower Grid Cylinder Guide Blocks

2.4 Age-related Degradation Mechanisms

The age-related degradation mechanisms considered in this report are:

1. Irradiation Embrittlement.
2. Stress Corrosion Cracking (SCC).
3. Irradiation-assisted Stress Corrosion Cracking (IASCC).
4. Erosion and Erosion/Corrosion.
5. Creep.
6. Stress Relaxation.
7. Wear.
8. Thermal Aging
9. Corrosion.
10. Fatigue.

These mechanisms were identified from a review of vendor-specific evaluations, nuclear power plant operation experience, relevant laboratory data, and related experience in other industries.

2.5 Evaluation Approach

The PWR RPV internals components that are within the scope of this report are evaluated with respect to the listed age-related degradation mechanisms in a four-step process. First, the evaluation basis is defined in Section 3 of this report by describing the component design features (Section 3.1), their design bases (Sections 3.2), and their operation and maintenance histories (Section 3.3). The latter includes an assessment of the generic applicability of reported service experience.

Second, the age-related degradation mechanisms are described and evaluated generically in Section 4 of the report with respect to their potential significance to the continued safety function performance of PWR RPV internals components during the license renewal term. An age-related degradation mechanism is defined to be significant for a component if, when allowed to continue without any additional prevention or mitigation measures, it cannot be shown that the component would continue to maintain its intended safety functions during the license renewal term. If it can be shown that the component is either not susceptible to the age-related degradation mechanism under consideration, or is susceptible to such a small degree that the component safety function performance is not affected throughout the license renewal term, then the component/degradation mechanism combination is not significant. Otherwise, the component/degradation mechanism combination is potentially significant.

Third, potentially significant component/degradation mechanism combinations are generically evaluated in Section 5 of the report with respect to the capability and corresponding basis of effective programs of inspection, testing, maintenance, surveillance, and analytical assessment to manage the effects of the age-related degradation. If a potentially significant

component/degradation mechanism combination is adequately addressed by effective programs, then the issue is considered to be resolved on the basis that the degradation is managed acceptably. Combinations of mechanisms and components for which it cannot be shown that effective programs manage potentially significant age-related degradation require plant-specific evaluation.

Fourth and finally, recommendations for plant-specific management options for significant age-related degradation are provided in Section 6 of this report.

2.6 License Renewal Applicant Responsibilities

A license renewal applicant intending to take credit for a particular conclusion of this report is responsible for the review and evaluation of plant-specific features, including appropriate CLB documents and information, to assure that the assumptions and criteria used as the basis for the determination of nonsignificance in Section 4 of this report are applicable to the component under consideration. Similarly, the license renewal applicant is responsible for the review and evaluation of their plant-specific features, including appropriate CLB documents and information, in order to assure that the program elements used to manage the effects of potentially significant degradation in Section 5 of this report, or their justified equivalent, are committed for use at their plant.

A license renewal applicant is responsible for performing plant-specific evaluations for any age-related degradation/component issues which the applicant is unable to demonstrate that the generic stipulated program elements are committed for use, or that the basis for the generic conclusions are applicable, at their plant. For those components within the scope of this report whose design basis includes allowance for functional deformation, the license renewal applicant is responsible for confirming the continued acceptibility of the limits established for this plant-specific functional deformation during the license renewal term. Finally, as a part of the review and evaluation of plant-specific features, including appropriate CLB documents and information, all license renewal applicants must re-evaluate time-dependent analyses, such as leak-before-break (LBB), to assure their validity for the license renewal term.

3.0 INTERNALS COMPONENTS EVALUATION BASIS

This section of the report provides the age-related degradation evaluation basis for specific PWR RPV internals components to support issuance of a renewed license. This evaluation basis consists of the design features of the components (Section 3.1), including design function, material specifications, and differences among the component designs of the three domestic PWR NSSS vendors; the codes, standards, and regulations that govern the construction and operation of the components (Section 3.2); and relevant items related to the operation and maintenance history of components (Section 3.3).

3.1 INTERNALS COMPONENTS DESIGN FEATURES

This section describes the evolution of pressurized water reactor internals from the earliest designs to the standardized designs currently in operation.

Descriptions of typical Westinghouse, Combustion Engineering and Babcock and Wilcox internals designs are presented in detail.

All PWR RPV internals consist of two basic elements: an upper internals package which is removed during each refueling operation in order to obtain access to the reactor core and a lower internals package which can be removed, if desired, following a complete core unload. The fact that all of the PWR internals can be removed from the reactor vessel ensures the ability to perform periodic inspections to determine the condition of the internals or to effect repairs, if needed. This unique characteristic of all PWR internals assures that the reactor internals will not preclude extended plant operation beyond the current licensed lifetime.

3.1.1 Design Evolution

Present-day commercial power reactors were preceded by several test/demonstration type plants. The first of these plants, Shippingport, went into service in the mid-1950's. It was followed into service by BR-3 in Belgium and Saxton in Pennsylvania around 1960. Although smaller in size, and different in detail, these PWRs had many similarities to present day designs. A fourth test/demonstration reactor was CVTR, a pressure tube reactor totally different from today's reactor plants and the only one of its kind. All of these plants have been decommissioned and, in fact, the dismantling of Shippingport has taken place.

Yankee Rowe was the first totally commercial pressurized water power reactor, and began commercial operation in June 1961. Two other Westinghouse plants of a similar style followed. These were Trino Vercellese in Italy and Sena-Chooz A in Belgium. These plants began commercial operation in the mid-1960s. In this same time frame Obrigheim, produced by Siemens, went on line in Germany. These plants can best be characterized as having a bladed "cruciform" shaped control rod with an attached "follower." The control rods operate

in slots formed by adjacent fuel assemblies. When the control rods are withdrawn, the "followers" fill the slots.

As these plants reached completion, today's basic design for pressurized water power reactors evolved. The most definitive characteristic of the most recent designs, which set them apart from the earlier plants, was the use of assemblies of cylindrical control rods, called rod control cluster assemblies (RCCA) by Westinghouse, control element assemblies (CEA) by Combustion Engineering, and control rod assemblies (CRA) by Babcock and Wilcox. These control assemblies operate within the fuel assemblies and eliminate the need for "followers" and the shroud assemblies which support them. The use of cylindrical control rods also eliminates the need to contour the fuel assembly and internals boundary shapes to accommodate the "cruciforms."

The resulting changes in fuel and control rod assembly design of the reactor internals system generally led to design simplification and efficiency. The subsequent evolution was fed by early government-sponsored programs, followed by extensive industry programs. Large test loops were utilized which simulated the flow conditions of operating plant reactors. Scale model tests, structural tests, materials tests, fabrication methods development programs and quality control procedures were initiated. The knowledge gained from these efforts, and data obtained from operating plants, were incorporated into operating plant design and analysis to produce the reactor internals systems in use today.

The first plant which used a cylindrical control rod assembly was San Onofre 1, followed by Connecticut Yankee (Haddam Neck), both of which began commercial operation in 1968. Hot functional or pre-operational testing of reactor internals systems began with these plants. This testing was used as a basis for Regulatory Guide 1.20, which requires vibration assessment testing of new PWRs.

Except for postulated accident conditions, the dominant design loads applied to PWR RPV internals result from flow induced vibration, thermal gradients, and postulated seismic events. Because of this, components were designed to have high stiffnesses while still maintaining sufficient flexibility to reduce thermal stresses. Evolutionary design changes in reactor internals designs have occurred as a result of increases in plant size, operating experience, fabrication techniques and the evolution of regulatory and code requirements. The design requirements and functions, as well as the overall design characteristics, have remained fundamentally the same. Section 3.1.2 provides descriptions primarily of the newer models. The important differences between these and earlier models are also noted.

3.1.2 Internals Components Descriptions

The purpose of this section is to provide the background necessary for a full understanding of the basic geometry of PWR RPV internals, their functional requirements, and assurance that these design requirements are satisfied. This description will concentrate on the generic aspects of reactor internals design. However, it must be recognized that there are, in some cases, features which are not common to all vendor designs or, for that matter, to all designs of a

given vendor. In these cases, specific design features will be identified. For commercial PWRs in the United States, there are three vendors: Westinghouse, Combustion Engineering, and Babcock & Wilcox.

Table 3.1 is a chronology of commercial PWRs in the USA which identifies reactor internals suppliers and operating license dates. Figures 3.2, 3.3, and 3.4 illustrate the typical features of Westinghouse, Combustion Engineering and Babcock & Wilcox PWR RPV internals. The reactor internals are designed to serve several functions:

1. Provide support and orientation of the reactor core.
2. Provide support, orientation, guidance and protection of the rod control cluster assemblies (RCCA) in Westinghouse plants. These are referred to in Combustion Engineering and Babcock & Wilcox plants, respectively, as control element assemblies (CEA), and control rod assemblies (CRA).
3. Provide a passageway for the distribution of the reactor coolant flow to the reactor core.
4. Provide a passageway for support, guidance, and protection for in-vessel/core instrumentation.
5. Provide a secondary core support for limiting the downward displacement of the core support structure in the event of a postulated failure of the core barrel.
6. Provide gamma and neutron shielding for the reactor vessel.
7. Provide positioning and support for the reactor vessel irradiation specimens (most CE plants excepted).

Commonalty exists in all PWRs in the overall means of structural support and guidance, and in the flow of coolant through the reactor vessel and internals. While there is considerable diversity in the particular components used to achieve core support, PWR RPV internals can generally be divided into two main subassemblies, a lower assembly which consists of a lower support structure attached to a core barrel, and an upper assembly which is installed after the lower assembly and fuel assemblies have been installed. The fuel assemblies rest on the lower support structure of the lower assembly, which transmits the resulting load to the core barrel and, thence, to the core barrel flange, which rests on the reactor vessel flange. The upper assembly is clamped under the reactor vessel head flange and provides the upper structure interface with the fuel assemblies. During refueling operations, the upper assembly is removed from the reactor vessel to allow access to the fuel assemblies. As noted earlier, this provides an excellent opportunity to perform inspections of the upper internals components on a regular basis.

The core barrel also provides a flow boundary for the reactor coolant as illustrated in Figure 3.1. When the primary coolant enters the reactor vessel, it impinges on the side of the core barrel and is directed downward through the annulus formed by the gap between the outside diameter of the core barrel and the inside diameter of the vessel. The flow then enters the

lower plenum area between the bottom of the lower support plate and the vessel bottom head and is redirected upward through the core. After passing through the core, the coolant enters the upper core support region and then proceeds radially outward through the reactor vessel outlet nozzles. The perforations in the various components, such as the lower support, control and distribute the flow to the core. In some reactor internals designs, a small amount of bypass flow is allowed to enter the vessel head plenum for cooling purposes.

For the evaluations in this report and as illustrated in Figure 3.1, the internals components are divided into three structural assembly groupings:

(a) the Upper Internals Assembly

(b) the Core Support Assembly

(c) the Lower Internals Assembly

The Upper Internals Assembly consists of all the internals components above the core which are removed at each refueling. For the purposes of this report, the remaining internals components have been arbitrarily divided into the Core Support Assembly and the Lower Internals Assembly, and none of these components are generally removed at refueling. The Lower Internals Assembly consists of all components below the core; the Core Support Assembly consists of the remaining components. The Upper Internals Assembly, Core Support Assembly and Lower Internals Assembly groupings are not necessarily the designations used by the three vendors; they are used here solely to provide a common framework for discussion.

Table 3.2 identifies the vendor specific designations of internals components covered by this report and contained within the Upper Internals, Core Support, and Lower Internals Assemblies. In Table 3.2, components which appear in the same row have the same general function in Westinghouse, Combustion Engineering, and Babcock and Wilcox internals designs. The absence of an entry means that a specific vendor design does not have such a component. For those components which have subcomponents which may be affected by specific age-related degradation mechanisms, these subcomponents are indented under the major component heading. Similarity in function does not extend to subcomponents which may appear on the same line.

Table 3.1
Chronology of U.S. Commercial Pressurized Water Reactors

KEY: (C) - ASME CODE-STAMPED INTERNALS
(U) - ASME CODE-STAMPED UPPER INTERNALS ONLY (REPLACEMENT)

PLANT	OWNER	REACTOR INTERNALS	NET MWe	OPERATING LICENSE
YANKEE ROWE	YANKEE ATOMIC ELECTRIC CO.	W	175	7/60
HADDAM NECK	CONNECTICUT YANKEE ATOMIC POWER CO.	W	582	6/67
SAN ONOFRE 1	SOUTHERN CAL EDISON & SAN DIEGO G & E	W	436	3/67
ROBERT E. GINNA	ROCHESTER GAS & ELECTRIC CORP.	W	490	9/69
POINT BEACH 1	WISCONSIN ELECTRIC POWER CO.	W	485	10/70
ROBINSON 2	CAROLINA POWER & LIGHT CO.	W	665	9/70
PALISADES	CONSUMERS POWER CO.	CE	777	10/72
POINT BEACH 2	WISCONSIN ELECTRIC POWER CO.	W	485	5/72
TURKEY POINT 3	FLORIDA POWER & LIGHT CO.	W	666	7/72
SURRY 1	VIRGINIA POWER CO.	W	781	5/72
MAINE YANKEE	MAINE YANKEE ATOMIC POWER CO.	CE	825	6/73
SURRY 2	VIRGINIA POWER CO.	W	781	1/73
OCONEE 1	DUKE POWER CO.	B&W	860	2/73
FORT CALHOUN 1	OMAHA PUBLIC POWER DISTRICT	CE	486	5/73
TURKEY POINT 4	FLORIDA POWER & LIGHT CO.	W	666	4/73
ZION 1	COMMONWEALTH EDISON CO.	W	1040	4/73
PRAIRIE ISLAND 1	NORTHERN STATES POWER CO.	W (U)	520	8/73
KEWANUNEE	WISCONSIN PUBLIC SERVICE CORP.	W	535	12/73
INDIAN POINT 2	CONSOLIDATED EDISON CO.	W	873	9/73
ZION 2	COMMONWEALTH EDISON CO.	W	1040	11/73
TMI1	GPU NUCLEAR CORP.	B&W	792	4/74
OCONEE 2	DUKE POWER CO.	B&W	860	10/73
ANO 1	ARKANSAS POWER & LIGHT CO.	B&W	836	5/74
OCONEE 3	DUKE POWER CO.	B&W	860	7/74
PRAIRIE ISLAND 2	NORTHERN STATES POWER CO.	W (U)	520	10/74
RANCHO SECO	SACRAMENTO MUNICIPAL UTILITY DIST.	B&W	913	8/74
CALVERT CLIFFS 1	BALTIMORE GAS & ELECTRIC CO.	CE	850	7/74
DONALD C. COOK	INDIANA & MICHIGAN ELECTRIC CO.	W	1020	10/74
MILLSTONE 2	NORTHEAST UTILITIES	CE	870	8/75
TROJAN	PORTLAND GENERAL ELECTRIC CO.	W	1130	11/75
INDIAN POINT 3	NEW YORK POWER AUTHORITY	W	965	12/75
ST. LUCIE 1	FLORIDA POWER & LIGHT CO.	CE	827	3/76
CRYSTAL RIVER 3	FLORIDA POWER CORP.	B&W	825	12/76
CALVERT CLIFFS 2	BALTIMORE GAS & ELECTRIC CO.	CE	850	8/76
BEAVER VALLEY 1	DUQUESNE LIGHT CO.	W	833	1/76
SALEM 1	PUBLIC SERVICE ELECTRIC & GAS CO.	W	1106	4/77
DAVIS-BESSE	TOLEDO EDISON CO.	B&W	866	4/77
FARLEY 1	ALABAMA POWER CO.	W	829	6/77
NORTH ANNA 1	VIRGINIA POWER CO.	W	893	11/77
DONALD C. COOK 2	INDIANA & MICHIGAN ELECTRIC CO.	W	1060	12/77
ANO 2	ARKANSAS POWER & LIGHT CO.	CE	858	7/78
NORTH ANNA 2	VIRGINIA POWER CO.	W	893	8/80

Table 3.1 (continued)
Chronology of U.S. Commercial Pressurized Water Reactors

KEY: (C) - ASME CODE-STAMPED INTERNALS
(U) - ASME CODE-STAMPED UPPER INTERNALS ONLY (REPLACEMENT)

PLANT	OWNER	REACTOR INTERNALS	NET MWe	OPERATING LICENSE
FARLEY 2	ALABAMA POWER CO.	W	829	10/80
SEQUOYAH 1	TENNESSEE VALLEY AUTHORITY	W	1148	9/80
SALEM 2	PUBLIC SERVICE ELECTRIC & GAS CO.	W	1106	8/81
McGUIRE 1	DUKE POWER CO.	W	1180	6/81
SEQUOYAH 2	TENNESSEE VALLEY AUTHORITY	W	1148	9/81
ST. LUCIE 2	FLORIDA POWER & LIGHT CO.	CE	837	4/83
SAN ONOFRE 2	SOUTHERN CAL EDISON & SAN DIEGO G & E	CE	1100	9/82
V. C. SUMMER	SOUTH CAROLINA ELECTRIC & GAS CO.	W	900	8/82
McGUIRE 2	DUKE POWER CO.	W	1180	3/83
SAN ONOFRE 3	SOUTHERN CAL EDISON & SAN DIEGO G & E	CE	1100	9/83
CALLAWAY 1	UNION ELECTRIC CO.	W (C)	1150	6/84
DIABLO CANYON 1	PACIFIC GAS & ELECTRIC CO.	W	1084	9/81
CATAWBA 1	DUKE POWER CO.	W	1145	6/85
BYRON 1	COMMONWEALTH EDISON CO.	W	1120	2/85
WOLF CREEK	WOLF CREEK NUCLEAR OPERATING CORP.	W (C)	1150	3/85
WATERFORD 3	LOUISIANA POWER & LIGHT CO.	CE	1104	12/84
PALO VERDE 1	ARIZONA PUBLIC SERVICE CO.	CE (C)	1270	12/84
DIABLO CANYON 2	PACIFIC GAS & ELECTRIC CO.	W	1106	8/85
MILLSTONE 3	NORTHEAST UTILITIES	W	1150	1/86
CATAWBA 2	DUKE POWER CO.	W	1145	5/86
PALO VERDE 2	ARIZONA PUBLIC SERVICE CO.	CE (C)	1270	12/85
SHEARON HARRIS	CAROLINA POWER & LIGHT CO.	W	900	1/87
VOGTLE 1	GEORGIA POWER CO.	W	1100	1/87
BYRON 2	COMMONWEALTH EDISON CO.	W	1120	1/87
BEAVER VALLEY 2	DUQUESNE LIGHT CO.	W	833	11/87
PALO VERDE 3	ARIZONA PUBLIC SERVICE CO.	CE (C)	1270	3/87
BRAIDWOOD 1	COMMONWEALTH EDISON CO.	W	1120	7/87
SOUTH TEXAS 1	HOUSTON LIGHTING & POWER CO.	W (C)	1250	8/88
BRAIDWOOD 2	COMMONWEALTH EDISON CO.	W	1120	10/88
VOGTLE 2	GEORGIA POWER CO.	W	1100	6/89
SOUTH TEXAS 2	HOUSTON LIGHTING & POWER CO.	W (C)	1250	6/89
COMANCHE PEAK 1	TEXAS UTILITIES ELECTRIC CO.	W	1150	4/90

Figure 3.1
PWR RPV Internals IR
Structural Assembly Groupings

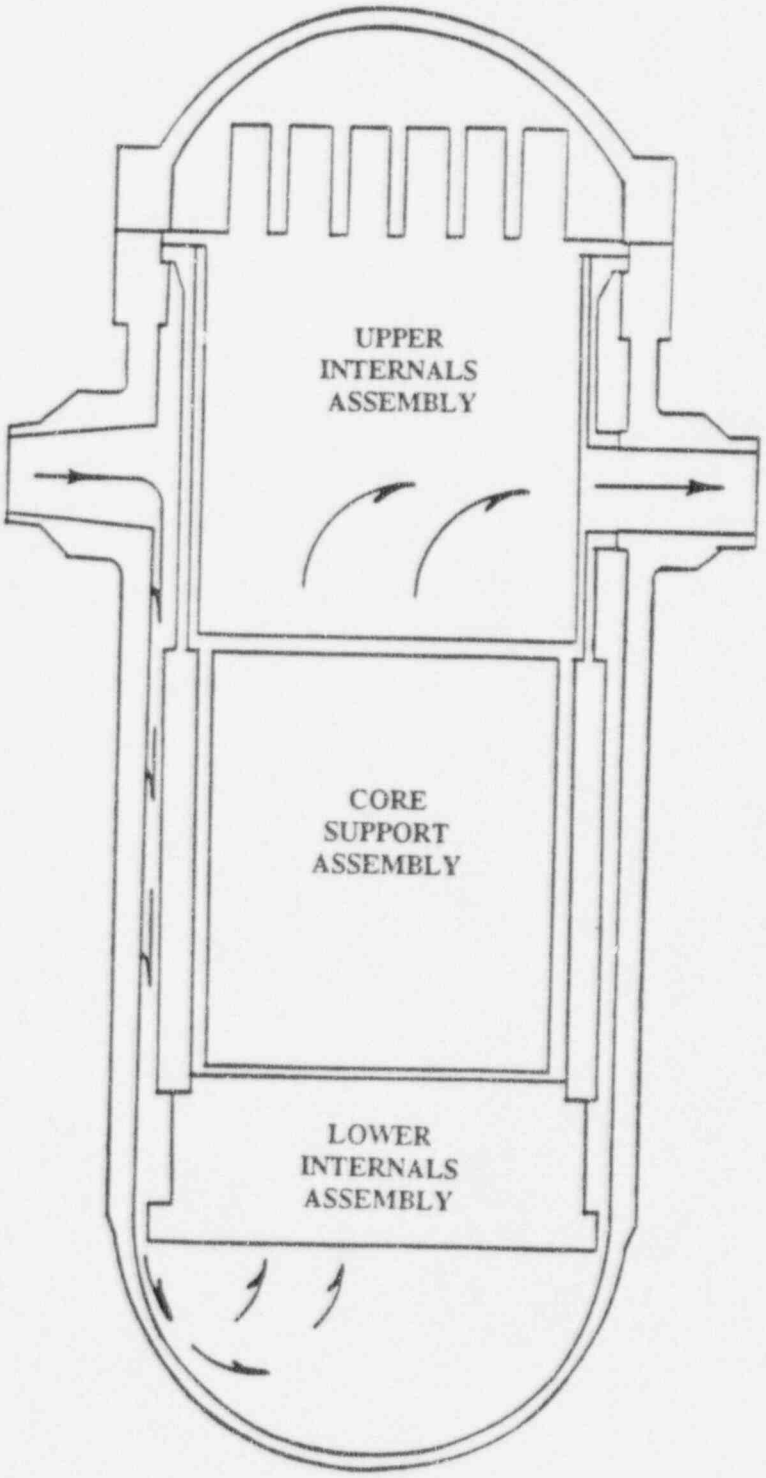


Table 3.2

VENDOR SPECIFIC COMPONENT DESIGNATIONS

for each STRUCTURAL ASSEMBLY GROUPING

Westinghouse	Combustion Engineering	Babcock & Wilcox
UPPER INTERNALS ASSEMBLY		
Upper Internals Assembly	**Upper Guide Structure Assembly**	**Plenum Assembly**
1. Upper Support Plate	1. Upper Guide Structure Support Plate	1. Plenum Cover and Plenum Cylinder
2. RCCA Guide Tube Assemblies RCCA Guide Tubes RCCA Guide Tube Bolts RCCA Guide Tube Support Pins	2. CEA Shroud Assemblies CEA Shrouds CEA Shroud Bolts	2. CRA Guide Tube Assemblies CRA Guide Tubes CRA Guide Tube Bolts
3. Upper Support Columns Upper Support Column Bolts	3. ----------	3. ----------
4. Upper Core Plate Fuel Pins	4. Fuel Alignment Plate	4. Upper Grid Assembly Upper Grid Rib Section Upper Grid Assembly Bolts Fuel Guide Pads
CORE SUPPORT ASSEMBLY		
Core Barrel and Baffle/Former Assembly	**Core Support Barrel and Core Shroud Assembly**	**Core Support Assembly**
1. Core Barrel Upper Core Barrel Flange Core Barrel Nozzles	1. Core Support Barrel Core Support Barrel Upper Flange	1. Core Support Shield Core Support Shield Flange
2. ----------	2. ----------	2. Vent Valve Assemblies
3. Baffle-Former Assembly Baffle / Former Assembly Bolts Baffle / Former Assembly Baffles	3. Core Shroud Assembly Core Shroud Assembly Bolts Core Shroud Tie Rods	3. Core Barrel Assembly Core Barrel Bolts Baffle / Former Plates Baffle / Former Bolts
4. Upper Core Plate Alignment Pins	4. Fuel Alignment Plate Guide Lugs	4. ----------
LOWER INTERNALS ASSEMBLY		
Lower Internals Assembly	**Core Support Assembly**	**Lower Grid Assembly**
1. Lower Core Plate	1. Core Support Plate	1. Lower Grid Top Rib Section
2. Fuel Pins	2. Fuel Alignment Pins	2. Fuel Guide Pads
3. Lower Support Plate	3. Lower Support Structure Beam Assemblies	3. Lower Grid Bottom Rib Weldment
4. Lower Support Columns Lower Support Column Bolts	4. Core Support Columns Core Support Column Bolts	4. Lower Grid Assembly Support Posts Lower Grid Assembly Bolts
5. Radial Keys and Clevis Inserts	5. Core Support Barrel Snubber Assemblies	5. Lower Grid Cylinder Guide Blocks

Table 3.3
PWR RPV Internals Materials of Construction

Material Category - Austenitic Stainless Steel

Material Type	Product Form	EXAMPLES[1] Components	Material Specifications
304 / 304L 316 / 316L 347	Bar	lugs, pads, pins, shims, plugs, retainers	A276 TP-304 A479 TP-304
	Plate	core barrel, plenum cylinder, former, baffle plates, guide tubes	A240 TP-304L A240 TP-304
	Forgings	flanges, nozzles, radial keys, lugs	A473 TP-304 A182 Gr F304
	Pipe/Tubing	pipe, support posts, column sleeves, column extensions	A312 TP-304 A316 TP-304 A213 TP-304
	Fastener	bolts, cap screws	A193 Gr B8 SA193 Gr B8M
308 / 308L	Welding Rod and Filler Metal	welds	A298 TE-308 A371 TER-308
Alloy A-286	Bar / Fastener	bolts, cap screws	SA-453 Gr 660
CF8 / CF3M	Casting	vent valve body, lower support columns	A351 Gr CF3M A351 Gr CF8

Material Category - Nickel-base Alloys

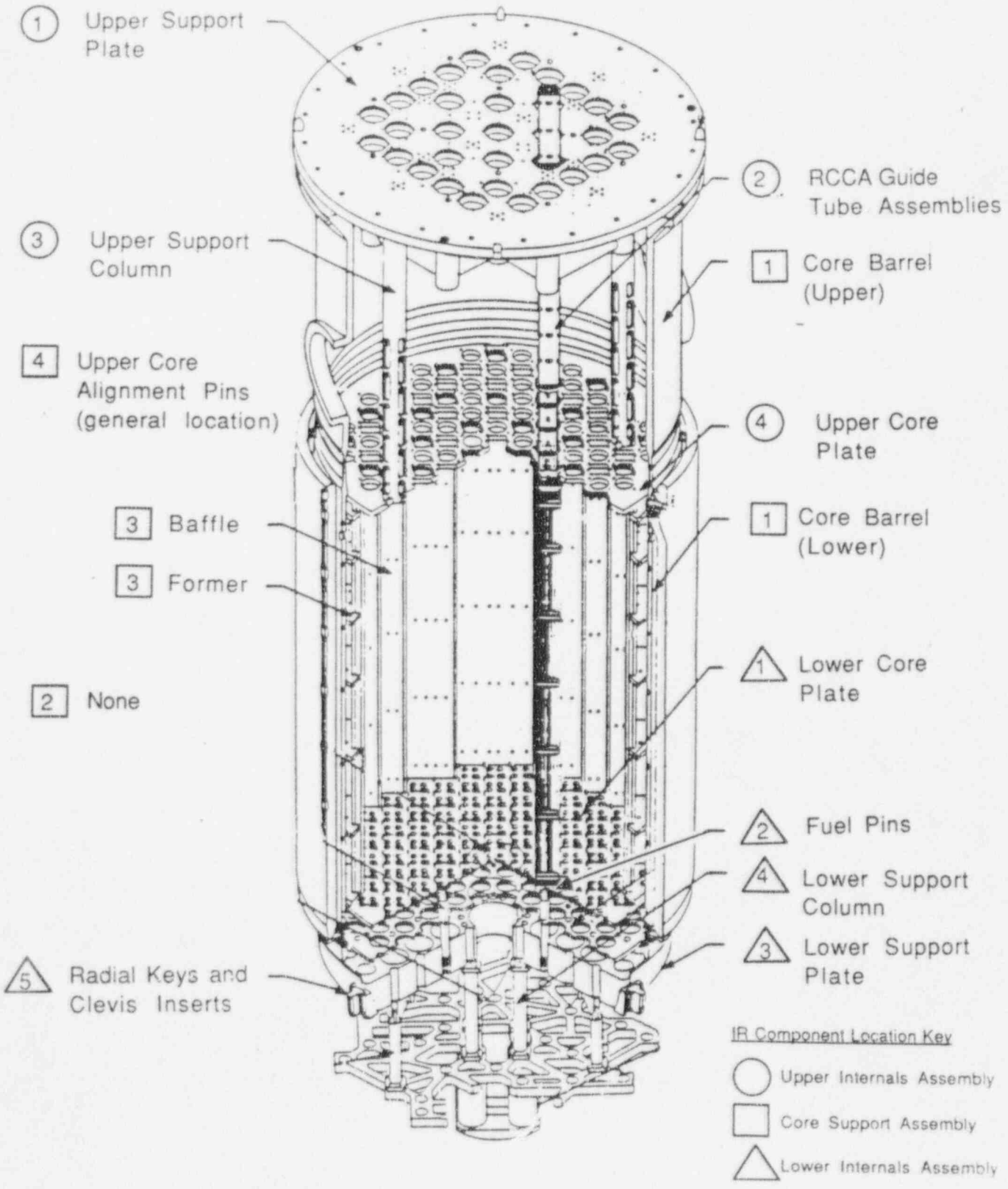
Material Type	Product Form	EXAMPLES[1] Components	Material Specifications
Alloy X-750	Bar / Fastener	studs, cap screws	SA-637
Alloy 600 [2]	Bar Plate	dowel pins, vent valve parts	B166 B168
Alloy 718 [2]	Bar / Fastener	cap screws	SA-637 Gr 718
Nicrobraze	Braze Metal	guide tube assembly brazing	80Ni-5Cr-4Fe-5Si-5B

NOTES:

(1) These are only representative examples of internals components constructed from the various materials specifications.

(2) Specific utility applications were identified. This report contains no generic evaluation of these materials.

Figure 3.2
Westinghouse RPV Internals System



3.1.2.1 Westinghouse Internals Component Descriptions

The Westinghouse components evaluated in each structural assembly group are described in this section. Figure 3.2 identifies each component and its relative location in the PWR RPV internals. The comparable components among the three NSSS vendor designs can be noted in Table 3.2.

3.1.2.1.1 Upper Internals Assembly

The upper internals assembly is located above the reactor core and is removed as a single component during refueling. The functions of the upper internals assembly are to align and support the upper ends of the fuel assemblies, maintain the spacing of the rod control cluster assembly (RCCA) guide tubes, prevent fuel assemblies from being lifted out of position during a severe accident condition, and protect the control rods from crossflow effects in the upper plenum.

Component One - Upper Support Plate The upper support plate provides core support and acts as a divider between the upper plenum and upper head, and also as a rigid base for the rest of the upper internals components. In Westinghouse plants, there are three styles of upper support plates: deep beam, top hat, and inverted top hat.

Component Two - RCCA Guide Tube Assemblies The rod control cluster assembly (RCCA) guide tubes provide the primary means of guidance for control rods. The RCCA guide tubes provide a straight, free, low friction path for the control rods, protect the control rods from the effects of crossflow in the outlet plenum, and provide storage for the control rod drive shafts during removal of the upper internals assembly. In most Westinghouse plants, a guide tube consists of an upper guide tube assembly above the upper support plate, and a lower guide tube assembly between the upper support plate and the upper core plate. Both assemblies are bolted to the upper support plate via a flange structure which aligns them with each other as well as attaching them to the upper support plate. At the upper core plate, the guide tubes are held in place by support pins which restrain guide tube lateral motion but provide no axial restraint. Openings are provided at the guide tube/upper core plate interface to allow passage of the reactor coolant flow to the outlet nozzles.

Component Three - Upper Support Columns The primary function of the upper support columns in Westinghouse plants is to transmit vertical loads from the core to the upper support structure, and is independent of any control rod support, guidance, or protection function. The support columns extend between the upper core plate and upper support plate, to which they are bolted. Combustion Engineering and Babcock and Wilcox internals do not have support columns, and rely on CEA shrouds and CRA guide tubes to serve the same function.

Component Four - Upper Core Plate The upper core plate in Westinghouse plants is a perforated plate bolted to the bottom of the upper support columns, which transfer vertical core loads to the upper support plate. The upper core plate compresses the fuel assembly hold-down springs, thus exerting a vertical preload on each fuel assembly. Lateral loads on

the upper core plate are reacted by the upper core plate alignment pins, which fit into keyways in the outer periphery of the upper core plate. The underside of the upper core plate contains the fuel pins which engage the top of the fuel assemblies. For plants with 14' diameter cores, the reverse is true: the top fuel pins are part of the fuel assembly and engage holes in the upper core plate.

3.1.2.1.2 Westinghouse Core Support Assembly

The core support assembly carries the static loads from the assembled components and fuel assemblies and dynamic loads from control rod trip, hydraulic flow, thermal expansion, seismic disturbances, and LOCA loads. The core support assembly and the lower internals assembly (Section 3.1.2.1.3) are contiguous parts of the internals assembly which provide core support and are generally not removed during refueling outages.

Component One - Core Barrel The core barrel is a flanged cylinder running the length of the reactor internals that mates to the reactor vessel at the top by the core barrel flange which rests on a ledge provided on the reactor vessel and at the bottom to the radial support system through the radial keys and clevis inserts. Loads are transferred from the core support and lower internals assemblies to the reactor vessel by the core barrel.

Component Two - None A comparable Westinghouse component to the B&W vent valves does not exist. Therefore, no evaluations of vent valve assemblies are required for the Westinghouse design.

Component Three - Baffle/former Assembly The baffle/former assembly consists of a structure of vertical baffle plates and horizontal former plates, which are bolted to the inside of the core barrel above the lower core plate. The baffle/former assembly provides an envelope for the core providing a coolant flow boundary which limits the amount of coolant bypass flow. It also limits the lateral motion of the fuel assemblies.

Component Four - Upper Core Plate Alignment Pins Upper core plate alignment pins located on the core barrel provide lateral core support during postulated accidents. In the Westinghouse design, peripheral pin/key arrangements at the core barrel inner diameter provide the interfaces between the core barrel and the upper internals assembly. The upper core plate alignment pins are on the core barrel in the Westinghouse design.

3.1.2.1.3 Westinghouse Lower Internals Assembly

The lower internals assembly positions and provides axial support for the core. The core is positioned inside the lower support assembly on top of the lower core plate of Westinghouse plants. The lower core plate is elevated above the lower support plate by support columns and bolted to a ring support attached to the inner diameter of the core barrel. The support columns transmit vertical fuel assembly loads from the lower core plate to the much thicker lower support plate. The lower support plate is welded to and supported by the core barrel, which transmits vertical loads to the vessel through the core barrel flange.

The guidance and alignment of the lower internals assembly during insertion into the reactor vessel is provided by the reactor vessel lower radial constraints and by the head/vessel alignment pins. The assembly is then supported by the core barrel flange which rests on the reactor vessel ledge and is clamped by the reactor vessel top head. Horizontal support of the lower internals assembly is provided by the flange to vessel interface, core barrel to reactor vessel outlet nozzles, and the radial keys and clevis inserts acting as lower radial constraints.

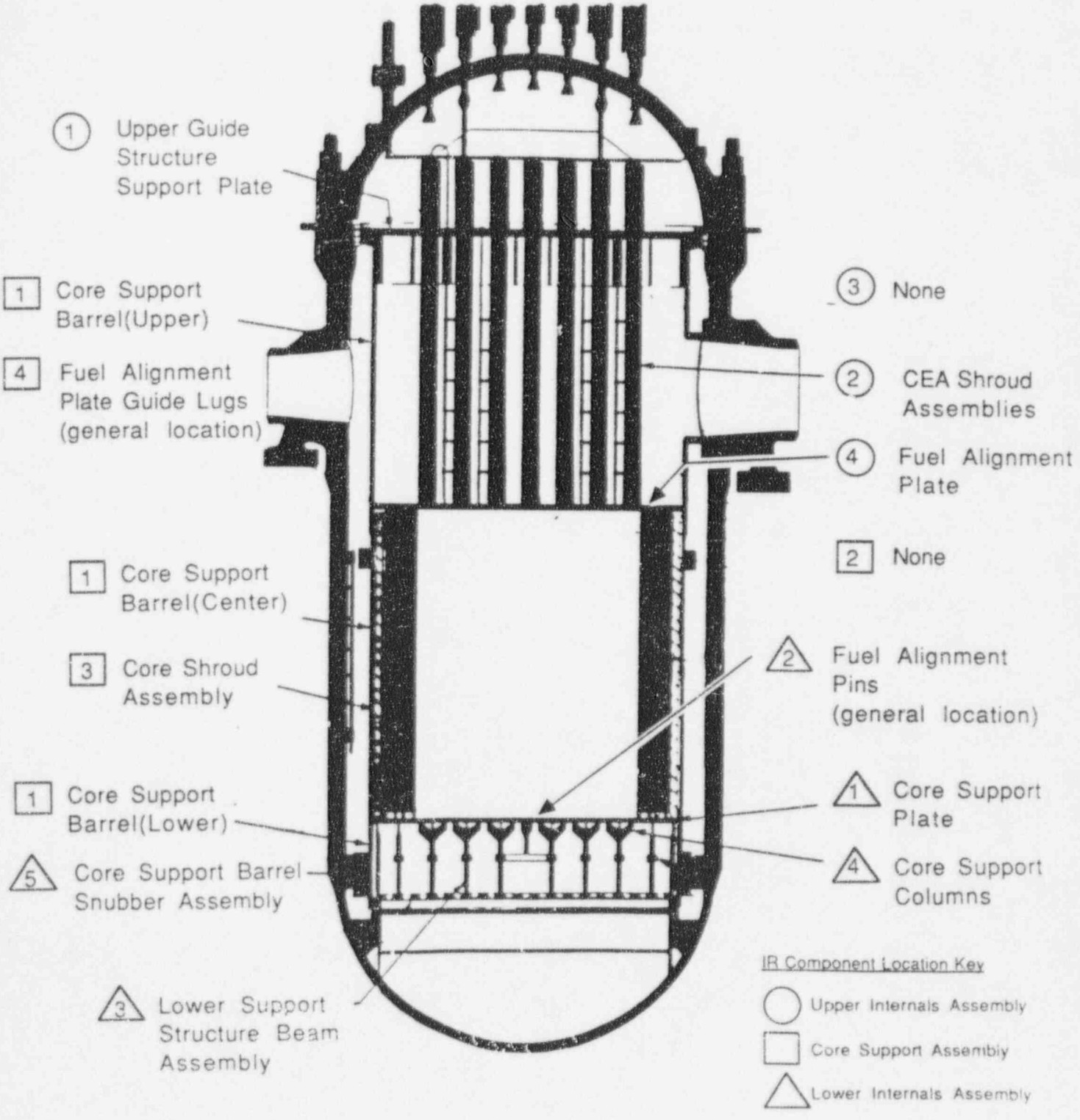
Components One and Two - Lower Core Plate and Fuel Pins The lower core plate and fuel pins in Westinghouse plants are core support structures and provide vertical and horizontal support for the fuel assemblies. The lower core plate positions and supports the reactor core and provides control of reactor coolant flow into each fuel assembly. It is located inside the core barrel and above the lower support plate. Fuel pins in the lower core plate provide guidance and lateral restraint for the individual fuel assemblies.

Component Three - Lower Support Plate The Westinghouse lower support plate provides core support by transmitting lower core plate loads through the lower support columns. It is attached with a full penetration weld to the lower end of the core barrel. The bottom mounted instrumentation assembly is attached to the underside of the lower support plate. In plants with 14' diameter cores, the lower support plate and lower core plate functions are combined into a single component and no lower support columns are used in the design.

Component Four - Lower Support Columns The lower support columns are core support structures and support the lower core plate and transmit the loads to the lower support plate. They are bolted at their top end to the lower core plate and at their bottom to the lower support plate structures. In plants with 14' diameter cores, no lower support columns are used in the design.

Component Five - Radial Keys and Clevis Inserts Radial keys and clevis inserts serve to limit lateral motion of the core barrel for normal operation and also during postulated accidents such as seismic and LOCA events. The lower core barrel in Westinghouse plants is restrained circumferentially by uniformly spaced radial keys. These keys, along with the matching clevis inserts attached to vessel-mounted clevis blocks, are designed to limit the lateral motion between the lower end of the core barrel and the reactor vessel. At assembly, as the internals are lowered into the vessel, the keys engage the keyways of the inserts in the axial direction. With the radial keys and clevis inserts, the radial and axial expansions of the core barrel are accommodated, but transverse static loads and vibrational movement of the core barrel are restricted.

Figure 3.3
Combustion Engineering RPV Internals System



3.1.2.2 Combustion Engineering Internals Component Descriptions

The Combustion Engineering components evaluated in each structural assembly group are described in this section. Figure 3.3 identifies each component and its relative location in the PWR RPV internals. The comparable components among the three NSSS vendor designs can be noted in Table 3.2.

3.1.2.2.1 Combustion Engineering Upper Internals Assembly

The upper internals assembly, known as the Upper Guide Structure Assembly in the CE design, is located above the reactor core and is removed as a single component during refueling. The functions of the upper internals assembly are to align and support the upper ends of the fuel assemblies, maintain the spacing of the control rod element assembly shrouds, prevent fuel assemblies from being lifted out of position during a severe accident condition, and protect the control rods from crossflow effects in the upper plenum.

Component One - Upper Guide Structure Support Plate The upper guide structure support plate provides core support and acts as a divider between the upper plenum and upper head, and also as a rigid base for the rest of the upper internals components. The upper guide structure support plate is welded to a grid array of deep beams and a deep cylinder which encloses and is welded to the ends of the beams.

Component Two - CEA Shrouds The control element assembly (CEA) shrouds provide the primary means of guidance for control rods. In addition, they also provide vertical core support in the CE design. The upper and lower sections of the CEA shrouds are welded together and are connected to the upper support plate with spanner nuts which permit axial adjustment. The shrouds are bolted to the fuel assembly alignment plate and transmit the core load to the upper support structure via the CEA shrouds. In some plants there are also dual-type shrouds which essentially consist of two single-type shrouds connected by a rectangular section shaped to accommodate dual control element assemblies. These dual shrouds are welded to the upper support plate. Openings are provided at the CEA shroud/fuel alignment plate interface to allow passage of the reactor coolant flow to the outlet nozzles.

Component Three - None A comparable Combustion Engineering component to the Westinghouse upper support columns does not exist. Therefore, no evaluations of upper support columns are required for the Combustion Engineering design.

Component Four - Fuel Alignment Plate The fuel alignment plate is designed to align the upper ends of the fuel assemblies and to support and align the lower ends of the CEA shrouds. The fuel alignment plate is a perforated plate bolted to the CEA shrouds. This plate compresses the fuel assembly hold down springs and exerts a vertical preload on each fuel assembly. Lateral restraint is provided by a slot/lug arrangement at the upper core plate/core barrel interface, much the same as in the Westinghouse key/pin arrangement (see Section 3.1.2.1.2). Fuel assembly alignment is provided by machined holes in the underside of the fuel alignment plate.

3.1.2.2.2 Combustion Engineering Core Support Assembly

The core support assembly carries the static loads from the assembled components and fuel assemblies and dynamic loads from control rod trip, hydraulic flow, thermal expansion, seismic disturbances, and LOCA loads. The core support assembly and the lower internals assembly (Section 3.1.2.2.3) are contiguous parts of the internals assembly which provide core support and are generally not removed during refueling outages.

Component One - Core Support Barrel The core support barrel is a cylindrically-shaped structure which envelopes the entire length of the RPV internals. It is suspended by a forged upper flange from a ledge on the reactor vessel and, in turn, supports the lower support structures. The core support barrel carries the static loads from the assembled structures and fuel assemblies and the dynamic loads from control rod trip, hydraulic flow, thermal expansion, seismic disturbances and LOCA loads.

Component Two - None A comparable Combustion Engineering component to the B&W vent valve assemblies does not exist. Therefore, no evaluations of vent valve assemblies are required for the Combustion Engineering design.

Component Three - Core Shroud Assembly The core shroud assembly provides an envelope for the core providing a coolant flow boundary which limits the amount of coolant bypass flow. It also limits the lateral motion of the fuel assemblies. The shroud consists of two stainless steel ring sections, aligned by means of radial shear pins and attached to the core support plate by stainless steel tie rods or, in later plants, by welds. A gap is maintained between the core shroud outer perimeter and the core barrel in order to provide some coolant flow between the core shroud and core support barrel, thereby minimizing thermal stresses in the core shroud and eliminating stagnant pockets. All Combustion Engineering plants have welded core shroud assemblies except for two earlier plant designs. In these two plants a bolted assembly, similar to Westinghouse and B&W plants, is used.

Component Four - Fuel Alignment Plate Guide Lugs Fuel Alignment Plate Guide Lugs, located on the core barrel, provide lateral core support during postulated accidents. In Westinghouse and Combustion Engineering plants, peripheral pin/key arrangements at the core barrel inner diameter provide the interfaces between the core barrel and the upper internals assembly. B&W designs do not use this particular pin/key arrangement.

3.1.2.2.3 Combustion Engineering Lower Internals Assembly

The lower internals assembly positions and provides axial support for the core. In Combustion Engineering plants, the lower end of the core support barrel is restrained by core support barrel snubber assemblies in contrast to the Westinghouse radial keys and clevis insert design. The fuel rests on the core support plate which is supported by core support columns resting on lower support structure beam assemblies. The beam assemblies thus serve the same function as the Westinghouse lower support plate.

Components One and Two - Core Support Plate and Fuel Alignment Pins

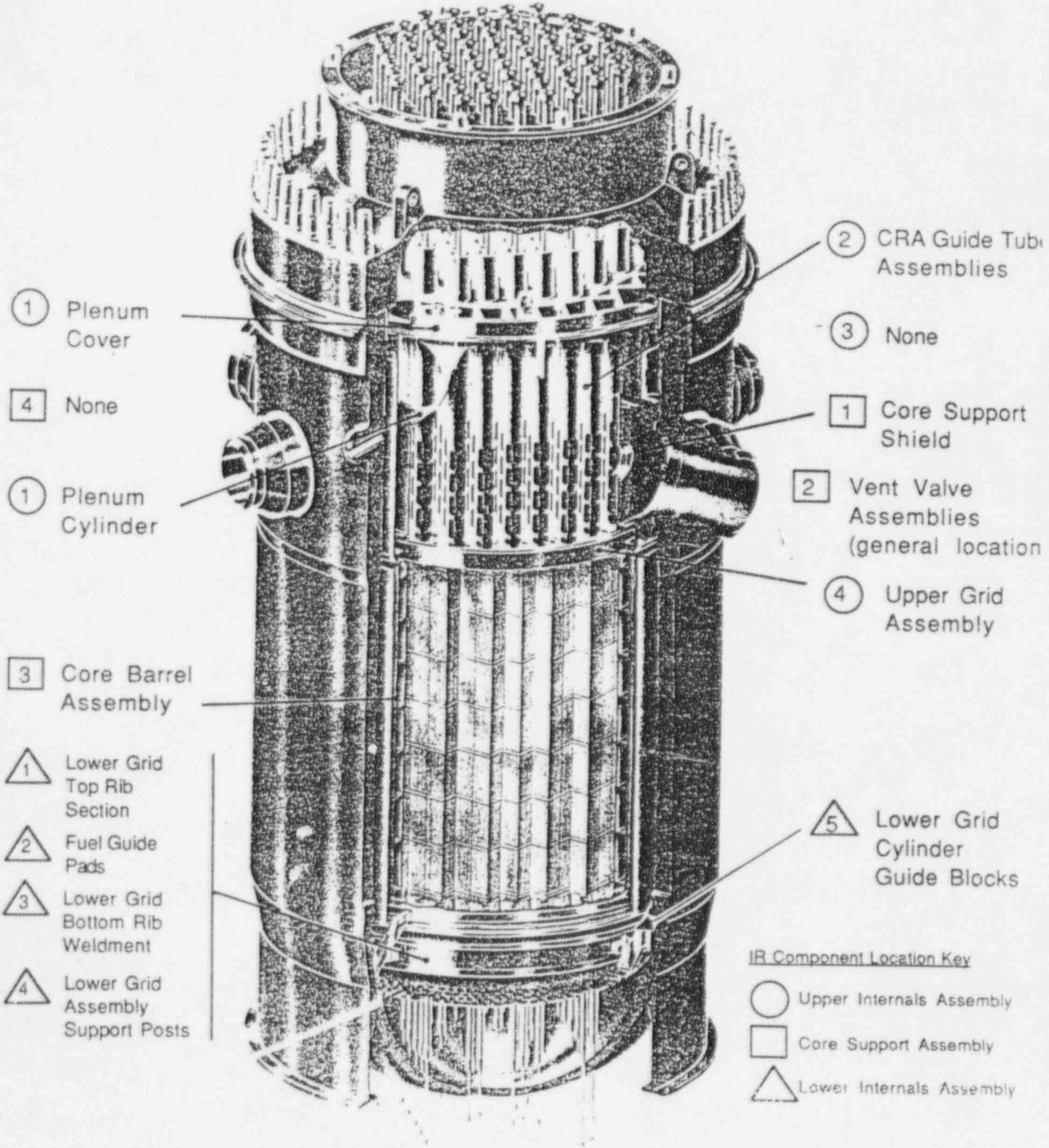
The core support plate and fuel alignment pins provide vertical and horizontal support for the fuel assemblies. The core support plate positions and supports the reactor core and provides control of reactor coolant flow into each fuel assembly. Fuel alignment pins, shrunk fit into the core support plate, provide guidance and lateral restraint for the individual fuel assemblies.

Component Three - Lower Support Structure Beam Assemblies Lower support structure beam assemblies are flanged to the bottom of the core support columns which, in turn, are attached to the core support plate. These assemblies allow core support plate loads to be transmitted within the core support structures.

Component Four - Core Support Columns The core support columns support the lower core plate and transmit the loads to the lower support structure beam assemblies. At their top, they are attached to the core support plate and at their lower end, they are flanged to the lower support structure beam assemblies.

Component Five - Core Support Barrel Snubber Assemblies Core support barrel snubber assemblies limit lateral motion of the core barrel for normal operation and also during postulated accidents such as seismic and LOCA events. Since the core support barrel is supported only at the upper end and the coolant flow could induce vibrations in the structure, snubber assemblies are installed on the outside of the core support barrel near the bottom end to limit lateral motion. The snubbers consist of equally spaced lugs around the circumference of the core support barrel and are the grooves of a "tongue-and-groove" assembly; the core stabilizing lugs attached to the pressure vessel are the tongues.

Figure 3.4
Babcock & Wilcox RPV Internals System



3.1.2.3 Babcock & Wilcox Internals Component Descriptions

The Babcock & Wilcox components evaluated in each structural assembly group are described in this section. Figure 3.4 identifies each component and its relative location in the PWR RPV internals. The comparable components among the three NSSS vendor designs can be noted in Table 3.2.

3.1.2.3.1 B&W Upper Internals Assembly

The upper internals assembly, known as the Plenum Assembly in the B&W design, is located above the reactor core and is removed as a single component during refueling. The functions of the upper internals assembly are to align and support the upper ends of the fuel assemblies, maintain the spacing of the control rod assembly guide tubes, prevent fuel assemblies from being lifted out of position during a severe accident condition, and protect the control rods from crossflow effects in the upper plenum.

Component One - Plenum Cover and Plenum Cylinder The plenum cover is constructed of a series of parallel flat plates intersecting to form a square lattice and has a perforated top plate with an integral flange at its periphery. The plenum cover assembly is flanged to the plenum cylinder top flange. The perforated top plate has matching holes to position the upper end of the CRA guide tubes. Locating keyways in the plenum cover flange engage the reactor vessel flange locating keys to align the plenum assembly with the reactor vessel, the reactor closure head CRD penetrations and the core support assembly (Section 3.1.2.3.2). The plenum cylinder consists of a large cylindrical section with flanges on both ends to connect to the plenum cover on the top end and the upper grid assembly on the lower end.

Component Two - CRA Guide Tube Assemblies The control rod assembly (CRA) guide tubes provide the primary means of guidance for control rods. In addition, they also provide vertical core support. The CRA guide tubes are welded to the plenum cover top plate and are bolted to the upper grid assembly. Each CRA guide tube consists of an outer tube housing, a mounting flange, twelve perforated slotted tubes, and four sets of tube segments which are oriented and attached (brazed) to a series of castings. The outer tube housing is welded to a mounting flange, which is bolted to the upper grid assembly. The CRA guide tubes also serve to transmit the core load to the plenum cover. Openings are provided at the guide tube/upper core plate interface to allow passage of the reactor coolant flow to the outlet nozzles.

Component Three - None A comparable B&W component to the Westinghouse upper support columns does not exist. Therefore, no evaluations of upper support columns are required for the B&W design.

Component Four - Upper Grid Assembly The upper grid assembly in Babcock & Wilcox plants consists of the upper grid rib section, a perforated plate which locates the lower ends of the CRA guide tube assemblies relative to the upper ends of the fuel assemblies, upper grid assembly bolts and fuel guide pads. The upper grid rib section compresses the fuel assembly hold-down springs, thus exerting a vertical preload on each fuel assembly. The upper grid rib

section is attached to the plenum cylinder lower flange by the upper grid assembly bolts. The underside of the upper grid rib section contains the fuel guide pads which engage the top of the fuel assemblies.

3.1.2.3.2 B&W Core Support Assembly

The core support assembly carries the static loads from the assembled components and fuel assemblies and dynamic loads from control rod trip, hydraulic flow, thermal expansion, seismic disturbances, and LOCA loads. The core support assembly and the lower internals assembly (Section 3.1.2.3.3) are contiguous parts of the internals assembly which provide core support and are generally not removed during refueling outages. In B&W plants the structure immediately adjacent to the core is called the core barrel assembly and this assembly is bolted.

Component One - Core Support Shield The core support shield is a flanged cylinder which mates with the reactor vessel opening. The forged top flange rests on a circumferential ledge in the reactor vessel closure flange. The core support shield lower flange is bolted to the core barrel. The inside surface of the lower flange guides and aligns the plenum cylinder relative to the core support shield. The cylinder wall has nozzle openings for coolant flow. These openings are formed by forged rings, which seal to the reactor vessel outlet nozzles by the differential thermal expansion between the stainless steel core support shield and the low alloy steel reactor vessel. The nozzle seal surfaces are finished and fitted to a predetermined cold gap, providing clearance for core support assembly installation and removal. At reactor operating temperature, the mating metal surfaces are in contact to make a seal without exceeding allowable stresses in either the reactor vessel or internals.

Component Two - Vent Valve Assemblies Vent valve assemblies are used to assist core cooling during a postulated loss of coolant event. In the typical B&W design, eight 14-inch inside diameter vent valve assemblies are installed in the cylindrical wall of the internals core support shield. Vent valves are designed to vent the steam generated within the reactor core, thereby assisting the rapid recovering of the core by coolant following a reactor coolant inlet pipe rupture. The valve centers are co-planar and are 42 inches above the plane of the reactor vessel coolant nozzle centers. In cross section, the valves are spaced around the circumference of the core support shield wall.

The hinge assembly of the vent valves 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 hinge journal contains integral exercise lugs for remote operation of the disc with the valve installed in the core support shield. The hinge assembly provides eight loose rotational clearances to minimize any possibility of impairment of free motion of the disc in service. In the event that one rotational clearance should bind in service, seven loose rotational clearances would remain to allow unhampered disc motion. In the worst case, at least four clearances must bind or seize solidly to adversely affect the valve disc motion.

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 external side of the disc is contoured to absorb the impact load of the disc on the reactor vessel inside wall without transmitting excessive impact loads to the hinge parts as a result of a loss-of-coolant accident.

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. In addition to the materials of construction listed in Table 3.3, the vent valve assemblies contain martensitic stainless steel (e.g., Types 15-5 PH and 431) used in several subcomponents and cobalt-base alloy (e.g., Stellite) used as a hardfacing material. Vent valve modifications to replace the locking devices were performed on four Oconee 1 vent valves in 1979. These changes resulted in additional nickel-base alloy materials being utilized in these Oconee vent valve assemblies including Alloy 600 and Alloy 718.

Component Three - Core Barrel Assembly The core barrel assembly in B&W plants supports the fuel assemblies, the lower grid, flow distributor, and in-core instrumentation guide tubes and restrains the core with respect to lateral and vertical motion. The core barrel consists of a cylinder which is flanged on both ends. The upper flange of the core barrel is bolted to the mating lower flange of the core support shield assembly and the lower flange is bolted to the lower grid assembly. Bolts are lock welded after final assembly (B&W Alloy A-286 replacement bolts utilize locking cups with multiple crimps and Alloy X-750 bolts also incorporate compression collars). Coolant flow is downward along the outside of the core barrel cylinder and upward through the fuel assemblies contained in the core barrel.

Component Four - None A comparable B&W component to the Westinghouse upper core plate alignment pins and the Combustion Engineering fuel alignment plate guide lugs does not exist. Therefore, no evaluations are required for the B&W design.

3.1.2.3.3 B&W Lower Internals Assembly

The lower internals assembly, known as the lower grid assembly in the B&W design, positions and provides axial support for the core. Components within this assembly are the lower grid top rib section, the lower grid bottom rib weldment, the lower grid assembly support posts, the lower grid cylinder and the lower grid cylinder guide blocks.

Components One and Two - Lower Grid Top Rib Section and Fuel Guide Pads The lower grid top rib section provides vertical and horizontal support for the fuel assemblies. The lower grid top rib section positions and supports the reactor core and provides control of reactor coolant flow into each fuel assembly. Fuel guide pads are provided in the lower grid assembly to provide guidance and lateral restraint for the individual fuel assemblies.

Component Three - Lower Grid Bottom Rib Weldment The B&W lower grid bottom rib weldment consists of intersecting plates welded to form a grid. Loads are transmitted to the lower grid bottom rib weldment from the lower internals structures through the lower grid assembly support posts.

Component Four - Lower Grid Assembly Support Posts The lower grid assembly support posts are core support structures and support the lower grid top rib section, transmitting loads to the lower grid bottom rib weldment. The lower grid assembly support posts are bolted at both ends.

Component Five - Lower Grid Cylinder Guide Blocks The lower grid cylinder is a large flanged cylinder which surrounds the assembly consisting of the lower grid top rib section, the attaching lower grid assembly support posts and the connected lower grid bottom rib weldment. The top flange of the forged lower grid cylinder is bolted to the lower flange of the core barrel. The guide blocks are attached to the lower grid cylinder which act as inserts to the guide lugs (or keys) welded to the reactor vessel inside wall. The elevation of these guide lugs is near the elevation of the vessel support skirt attachment which minimizes dynamic loading effects on the vessel shell or bottom head. The guide blocks limit lateral motion of the core barrel for normal operation and also during postulated accidents such as seismic and LOCA events.

3.2 INTERNALS COMPONENTS CODE, STANDARD AND REGULATION BASES

3.2.1 General Design Background

Before the development of ASME Code requirements specifically applicable to reactor internals, the design of reactor internals was based on criteria specific to each vendor. However, Section III of the ASME Boiler and Pressure Vessel Code, Subsection NB was used as a guideline for the development and establishment of vendor-specific internals system design criteria, and ultimately for the development of Subsection NG. A subsequent development was the evaluation of all reactor internals systems for seismic loading and the asymmetric loads resulting from a postulated pipe rupture.

PWR internals whose contract dates follow the issuance of the 1974 Edition of the ASME Boiler and Pressure Vessel Code Section III, are designed to satisfy Subsection NG, Core Support Structures. Among the requirements contained in Subsection NG are rules for fatigue evaluation and categorization of internals loads. The rules for elevated temperature service of metals whose temperatures exceed the ASME Section III allowables are in Code Case N-201 [3-1].

For design purposes, Subsection NG divides the individual reactor internals components into two categories:

a. Core support structures
b. Internals structures

Core support and internals structures consist of all structures within the reactor vessel except fuel assemblies, control assemblies, and instrumentation. The core support structures are comprised of those components whose function is to support and restrain the core. The components comprising the internal structures generally serve individual nonsupport functions necessary to the proper operation of the reactor plant, or serve a core support or restraint function only after a postulated failure of core support structures.

This categorization is made in order to distinguish between those structures which must satisfy the most stringent design criteria (core support structures), and those for whom the application of less demanding criteria may be tolerable (internals structures). It has been industry practice to apply the more stringent core support structure criteria to many of the internal structure components, particularly if they support a safety function. Core support structures, when designed according to Subsection NG, are referred to as Code Class CS, and receive a code stamp. Table 3.1 provides an identification of which PWR RPV internals have been code-stamped.

3.2.2 Internals Load Categorization

The reactor internals systems are designed to withstand steady state and fluctuating forces produced under handling, normal operating, transient and accident conditions. The types of

forces considered are mechanical, hydraulic, and thermal, and include deadweight, handling, preloads, temperature differential, system pressure, flow-induced vibration loads, seismic loads, and equipment operating loads. The stresses resulting from these loads have been determined analytically, or in some cases, experimentally. The stresses are categorized and compared against established allowable stresses. The ASME Code refers to this categorization as a "Hopper Diagram."

In the 1974 edition of Subsection NG, there are four categories entitled: 1) normal conditions; 2) upset conditions; 3) emergency conditions; and 4) faulted conditions. Later Code additions clarified this nomenclature but basically retained the same stress allowables. The corresponding new categories are: 1) Service Level A; 2) Service Level B; 3) Service Level C; and 4) Service Level D.

In addition, there are design requirements imposed by the reactor internals system supplier such as limiting deformation of various components. These requirements are usually stated in the supplier's design specification and design report. For pre-1974 plants, the supplier design specification and design reports covered these requirements. In addition, older plants were reviewed under the Integrated Plant Safety Assessment Program [3-2].

The following ASME Code limitations on stresses or deformations are employed to ensure capability of a safe and orderly shutdown in the event of earthquake and major loss-of-coolant incident loading conditions:

(a) Under normal operating loading plus operating basis earthquake (OBE) forces, the critical reactor vessel internal structures are designed within the stress criteria established in ASME Section III, Subsection NG.

(b) Under normal operating loadings plus safe shutdown earthquake (SSE) forces, the design criteria permit a small amount of local yielding.

(c) Under normal operating loadings plus reactor coolant pipe rupture loadings plus safe shutdown earthquake (SSE) forces, permanent deformation is permitted by the design criteria.

3.2.3 System Hot Functional Test

As a final step in the internals evaluation for service, pre-operational testing at hot conditions is performed. Nuclear Regulatory Guide 1.20 [3-3] establishes guidelines for the preoperational assessment program. Prior to the issuance of Regulatory Guide 1.20, hot functional testing was also performed by the industry. This testing formed the basis for establishing Regulatory Guide 1.20.

The system hot functional test is done at the plant site and follows the primary system hydrostatic test. The internals are installed in the reactor vessel without fuel and control rods. Prototypic reactor coolant temperatures are established by pump heating.

There are two basic levels of the hot functional test: (1) Prototype or first of a kind; and (2) Non-prototype or a repeat model. The first is a more extensive test and requires instrumentation of selected components to measure the structural response to hydraulic forces. Prior to the test, predictions of the response are established by the designers. Upon completion of the test, the data is evaluated to ensure compliance with the acceptance criteria. In addition, a detailed inspection of the internals system is conducted with the reactor internals removed from the reactor vessel. The inspection includes corresponding support areas of the reactor vessel. The non-prototype test is the same as the prototype, with the exception that it does not include the instrumented portion. However, a detailed inspection of the internals is again performed.

Hot functional tests are normally conducted once during plant life, prior to loading the first core. Phases of the test include operation with one or more coolant pumps. The test is conducted at elevated temperature and at a flow rate which is greater than during normal plant operation, due to the absence of fuel assembly flow resistance. The primary portion of the test consists of operation with all loop pumps for a minimum total period of ten days. Operation for such periods of time insures that flow-induced load cycling of internals components will be well into the high cycle range of their material fatigue design curves, thus providing assurance that component high cycle fatigue usages will be low throughout the plant design life.

3.2.4 Inservice Inspection (ISI)

Inservice inspection (ISI) is conducted at ten year intervals, with the option of deferral to the end of the current operating interval. Utilities are given the option of choosing one of two inspection program schedules, A or B, during each ten year interval. Program A is based on a greater concentration during the early years of operation, while Program B is based on a temporally uniform inspection program. For example, Program A requires 100% of the mandated inspections to be complete by the end of the third year of operation; Program B requires a minimum of 16% at the end of the third year, 50% at the end of the seventh year, and 100% at the end of the tenth year. The internals, with the exception of components attached to the vessel, are completely removed from the reactor vessel for the examination at least once during each interval. Inservice inspection requirements for PWR RPV internals components are given in Subsection IWB (Requirements for Class 1 Components of Light-Water Cooled Power Plants) of ASME Section XI. Table IWB-2500-1 of Section XI identifies PWR RPV internals as removable core structures whose accessible surfaces require visual inspection (VT-3) as discussed below:

(a) Determine the general mechanical and structural condition of components and their supports, such as the verification of clearances, settings, physical displacements, loose or missing parts, debris, corrosion, wear, erosion, or the loss of integrity at bolted or welded connections.

(b) Include examinations for conditions that could affect operability or functional adequacy of snubbers, and constant load and spring type supports.

(c) For component supports and component interiors, the visual examination may be performed remotely with or without optical aids to verify the structural integrity of the component.

In order to further clarify and standardize these requirements, the ASME has initiated the preparation of more definitive requirements which will ultimately be included in Section IWG of ASME Section XI.

3.2.5 Regulations and Regulatory Guidance Relevant to Internals Components

Additional regulatory guides and bulletins in existence are relevant to internals component aging degradation and its management. These are discussed in the subsections below.

Adoption of "Leak-Before-Break" in 10CFR50, Appendix A, General Design Criteria 4 (GDC-4), "Protection Against Dynamic Effects of Postulated Pipe Rupture" [3-4] - This document contains the requirements for the analysis of loads which result from postulated pipe ruptures. While these requirements do not directly affect license renewal, they may become relevant if internals degradation is postulated which may question critical assumptions of the design basis LOCA analysis, assuming the original design criteria include this requirement.

The leak-before-break technology has been accepted by the Nuclear Regulatory Commission (1987-1988) and has been incorporated into GDC-4. Recommendations on the application of the leak-before-break approach have been provided by the NRC in NUREG 1061, Volume 3 [3-5]. This technology provides a basis whereby utilities can eliminate postulated pipe ruptures from consideration if it can be shown, through the application of fracture mechanics analysis, that such breaks cannot occur. This technology applies to all high-energy piping, including the primary reactor coolant system piping. Its adoption provides one means of addressing safety-related internals aging degradation relevant to the LOCA event.

NRC Regulatory Guide 1.97, "Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant and Environs Conditions During and Following an Accident" [3-6] - This regulatory guide states the following: "... the required instrumentation (for monitoring accidents such as LOCA and seismic events) be capable of surviving the accident environment in which it is located for the length of time its function is required. Therefore, it could either be designed to withstand the accident environment or be protected by a local protected environment."

The particular instruments which are designated to satisfy this requirement are identified in the technical specifications for each plant. The requirement applies to those RPV internals components which are necessary for ensuring that the designated instrumentation not lose its safety function during and following such accidents. Typically, internals components which are used for this purpose are designated as internals structures, as defined in Section 3.2.1.

Regulatory Guide 1.97 is summarized here because of its relevance to the identification of safety issues potentially resulting from internals component degradations which can affect instrumentation integrity.

3.3 OPERATION AND MAINTENANCE HISTORY

The determination of which age-related degradation mechanisms were to be evaluated in the PWR RPV internals IR was based both upon the degradation that has actually occurred as well as that which potentially could cause internals component degradation. This subsection describes observations of PWR RPV internals age-related degradations that have occurred in the past and may be relevant to extended plant operation.

The history of commercial pressurized water reactor internals in the United States is one of safe, relatively trouble-free operation. There are no instances in which RPV internals degradation has resulted in a threat to public safety. In those cases where degradation has occurred, it has been identified, evaluated and, if necessary, repaired well before any potential safety issue could have arisen.

Nevertheless, there have been instances of internals degradation which can be considered significant. Even though these cases of degradation have not been permitted to develop to the point of becoming actual safety issues, some have raised questions relative to continued operation. Therefore, a historical review of the degradation mechanisms to which RPV internals have shown a consistent degree of vulnerability is an important requisite to addressing the outstanding issues confronting license renewal. The aim of this subsection is to summarize the issues which have affected operating pressurized water reactor internals. This will be done according to the structural assembly groupings of Section 3.1.

3.3.1 Upper Internals Assembly

Westinghouse Guide Tube Support Fasteners - The only significant generic issue associated with control rod guide tubes concerns the Alloy X-750 support pins which provide lateral support for the guide tubes at the upper core plate in most Westinghouse plants [3-7 through 3-10]. Some of these pins have experienced stress corrosion cracking.

The support pin degradation issue has been addressed on a plant-specific basis either by complete support pin replacement or by performing inspections which amply demonstrate that support pin degradation has not taken place. The replacement support pin design also employs Alloy X-750 material, but with a different heat treatment which is substantially less prone to stress corrosion cracking in the PWR environment. At San Onofre 1, inspections have demonstrated that degradation of the original support pins has not taken place in 21 years of service, and support pin replacement at this plant has, therefore, not been necessary. For all of the plants potentially affected, no further indications of degradation have occurred since the replacement/inspection programs initiated to address the issue have been in effect [3-7 through 3-11].

3.3.2 Core Support Assembly

B&W Core Barrel Bolts - The Alloy A-286 bolt aging degradation issue in B&W plants which is discussed more thoroughly in Section 3.3.3 below also applies to the bolts fastening the core support shield and core barrel in the core support assembly. The plant-specific programs discussed in Section 3.3.3 have assured that no further recurrence of the observed degradations have occurred to date [3-7, 3-8, 3-12 through 3-16].

Combustion Engineering Core Barrel - At two Combustion Engineering plants, flow-induced vibration of the thermal shields has resulted in damage to the base metal and welds of the thermal shields support lugs attached to the core barrel [3-7]. The thermal shields and support lugs were ultimately removed from these two plants to eliminate the problem. Damage to the core barrels was corrected by machining surface flaws and by drilling crack arrestors at the ends of the through-wall cracks present. Programs to evaluate the effects of these modifications on plant safety and performance have been conducted, with favorable results. Monitoring, inspection, maintenance, and in some cases, modification programs have also been instituted at other CE plants which could be potentially affected, to preclude the possibility of significant degradation. These programs have been successful in preventing degradation of the type experienced. In practice, this damage was caused by high-cycle fatigue which, as expected, manifested itself early in plant operating life and which is not a concern during an extended operating period.

Foreign Plant Baffle Bolts - In domestic plants, there have been no historical incidents which have involved degradation of the bolts which attach the baffle plates to the former plates in Westinghouse, B&W, and the few Combustion Engineering plants that use bolts. There have been recorded incidents of aging degradation at foreign plants built by Siemens and Kraftwerke Union in the 1986-1987 time frame [3-17]. The plants which have been affected are Biblis A and B, Goesgen, Neckarwestheim, and Unterweser. The aging degradation was due to stress corrosion cracking, caused by lockwelding of the Alloy X-750 bolts. The affected fasteners were replaced, and were mechanically locked in place. No further incidents have been reported. Recent inspections at FRAMATOME plants [3-18] have also identified failure of a small percentage of Type 316 stainless steel baffle-former bolts.

3.3.3 Lower Internals Assembly

B&W Lower Grid Assembly Bolts - The lower internals assemblies of most operating plants have been free of significant aging degradation. The only generic incidents of degradation in domestic plants have been to the B&W lower thermal shield and lower core barrel bolts made from Alloy A-286 which attach the thermal shield and the core barrel to the lower grid assembly [3-7, 3-8, 3-12 through 3-16]. Stress corrosion cracking of these bolts and of other internals bolts made of the same material has taken place. Subsequent examinations of degraded bolts have shown the stress corrosion to be intergranular in nature.

The issues arising from these aging degradations have been addressed by a variety of plant-specific approaches. Bolts which were inspected and found to be degraded have been replaced with bolts made of alternate designs of the same material in which stresses were reduced, or of an alternate high strength material (Alloy X-750). Bolts not found to be in a degraded condition have been allowed to remain in place. The programs initiated by each of the affected utilities have not identified any recurrence of bolt aging degradation to date [3-7, 3-8, 3-12 through 3-16].

Combustion Engineering Snubber Wear - A visual inspection in 1973 [3-7 and 3-13] revealed worn areas in the reactor vessel flange and head resulting from holddown spring degradation and subsequent internals vibration. Prior to shutdown, larger than normal ex-core neutron detector readings had suggested the possibility of excessive internals vibration. Wear was found on the mating surfaces, alignment keys and slots, snubbers, and outlet nozzle faces.

The worn surfaces were repaired, and a new design using Belleville spring assemblies has greatly increased holddown capacity. No subsequent degradation of this type has been observed.

3.4 SECTION 3 REFERENCES

[3-1] Code Case N-201: "Class CS Components in Elevated Temperature Service," Modification of Subsection NG, Section III, of ASME Boiler and Pressure Vessel Code.

[3-2] NUREG-0825, "Integrated Plant Safety Assessment, Systematic Evaluation Program," June 1983.

[3-3] Nuclear Regulatory Guide 1.20, "Comprehensive Vibration Assessment Program for Reactor Internals During Preoperational and Initial Startup Testing," December 1970.

[3-4] Nuclear Regulatory Commission 10 CFR Part 50, Appendix A, General Design Criterion 4, "Protection Against Dynamic Effects of Postulated Pipe Rupture."

[3-5] Nuclear Regulatory Commission NUREG-1061, Volume 3, "Report of the U.S. Nuclear Regulatory Commission Piping Review Committee - Evaluation of Potential for Pipe Breaks," November 1984.

[3-6] NRC Regulatory Guide 1.97, "Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant and Environs Conditions During and Following and Accident," Rev. 3, May 1983.

[3-7] S. M. Stoller Corporation, Nuclear Power Experience, January 1973 to January 1989

[3-8] Koo, W. H., "Threaded Fastener Experience in Nuclear Power Plants," NUREG-0943, January 1983.

[3-9] Mager, T. R., "Analysis of Failures in Ni-Cr-Fe Alloy X750 Internals in Westinghouse Plants," presented at the Bolt Degradation or Failure in Nuclear Plants Seminar, Knoxville, Tennessee, November 2-4, 1983, Revision 1, December, 1983.

[3-10] "Four Years On, EdF PWR's Again Suffer Pin Cracking," Nucleonics Week, March 17, 1988, p. 3.

[3-11] SAP-89-193, "Guide Tube Support Pin Update," Westinghouse Electric Corporation, July 5, 1989.

[3-12] S. M. Stoller Corporation, "Nuclear Unit Operating Experience, 1980 through 1982 Update," EPRI HP-Report NP-3480, April 1984.

[3-13] "Characterization of the Performance of Major LWR Components," EPRI Report NP-5001, January 1987.

[3-14] S. M. Stoller Corporation, "Nuclear Unit Operating Experience; 1978 and 1979 Update," EPRI Report NP-2092, October 1981.

[3-15] Moore, K. E., "Steam Generator Manway and Reactor Vessel Internals Fastener Experience in Babcock & Wilcox Plants," presented at the Bolt Degradation or Failure in Nuclear Plants Seminar, Knoxville, Tennessee, November 2-4, 1983, Revision 1, December 1983.

[3-16] R. S. Piascik et al., "Stress Corrosion Cracking of Alloy A-286 Bolt Material in Simulated PWR Reactor Environment," Proceedings of the Second International Symposium on Environmental Degradation of Materials in Nuclear Power Systems--Water Reactors, Monterey, California, September 9-12, 1985, pp. 18-25.

[3-17] "Screw Replacements in KWU-built PWR Cores to Continue," Nucleonics Week, March 5, 1987.

[3-18] "Baffle Assembly Screw Damage," INPO Nuclear Network Significant Event Report, November 17, 1989.

4.0 AGE-RELATED DEGRADATION MECHANISM ASSESSMENT

In the previous section of the report, the component evaluation bases for specific PWR RPV internals components were described, consisting of component design features (Section 3.1); codes, standards and regulations governing their construction and operation (Section 3.2); and relevant operation and maintenance histories (Section 3.3). This section of the report describes the age-related degradation mechanisms that could affect these specific PWR RPV internals components and evaluates the potential significance of the effects of these mechanisms on the continued safety function(s) performance of these components throughout the license renewal term.

The set of age-related degradation mechanisms evaluated in this section is derived from the available literature, in particular the Westinghouse [4-1], Combustion Engineering [4-2], and Babcock and Wilcox [4-3] plant-specific reports on PWR RPV internals as well as a review/evaluation of component service experience, relevant laboratory data, and related experience from other industries [4-4]. The set consists of the following:

1. Irradiation Embrittlement.
2. Stress Corrosion Cracking (SCC).
3. Irradiation-assisted Stress Corrosion Cracking (IASCC).
4. Erosion and Erosion/Corrosion.
5. Creep.
6. Stress Relaxation.
7. Wear.
8. Thermal Aging
9. Corrosion.
10. Fatigue.

When experience has shown combining two or more of these mechanisms to be significant, any such synergistic effect has been explicitly evaluated.

The technical evaluation of a particular age-related degradation mechanism and its effects on the continued safety performance of a particular PWR RPV internals component leads to one of two conclusions: (1) the effects of the mechanism are potentially significant to that component, and further evaluation is required in Section 5 relative to the capability of effective programs to manage the effects of the age-related degradation; or (2) the effects of the age-related degradation are not significant to the ability of that component to perform its intended safety function throughout the license renewal term. For the latter case, specific criteria and corresponding justification are provided in this section that can be used as the basis for generic resolution of the age-related degradation mechanism/component issue.

License renewal applicants intending to reference these generic conclusions are responsible for a review/evaluation of plant-specific features, including appropriate current licensing basis

(CLB) documents/information, in order to assure that there are no deviations from the assumptions and criteria used in the report. This review should compare the design basis for particular components with the representative design bases given in Sections 3.1 and 3.2. The component operation and maintenance histories should also be compared to the generic performance parameters described in Section 3.3. Finally, the specific assumptions and criteria used in this section should be examined to assure that they, or justified equivalents, apply to the component under consideration.

4.1 IRRADIATION EMBRITTLEMENT

4.1.1 Mechanism Description

Exposure to high energy neutrons (neutron energies greater than 0.1 MeV) can cause changes in the properties of the stainless steel and nickel-base alloys used in reactor internals. This neutron irradiation can produce changes in mechanical properties by increasing yield and ultimate stresses and correspondingly decreasing ductility and fracture toughness of internals component materials. The extent of irradiation embrittlement is a function of both the irradiation temperature and the neutron fluence. The nominal irradiation temperature for reactor internals is determined by the primary coolant temperature (500-650°F) and local gamma heating rates.

Data from low temperature thermal reactor irradiation of Type 304 and Type 316 stainless steel are available from several studies [4-5, 4-6 and 4-7]. Embrittlement, as evidenced by increases in yield stress and decreases in uniform and total elongation, is common in these materials after irradiation. Studies [4-5 and 4-6] have shown that embrittlement of stainless steel occurs at fluences as low as 1 x 10^{20} n/cm^2 (E > 1 MeV). These same studies have shown that the rate of change in mechanical properties is reduced at fluences above 2 x 10^{22} n/cm^2 (E > 1 MeV).

Programs have been established to determine the properties of materials exposed to irradiation in operating PWRs. Experimental data shows that the following properties result after prolonged exposure to neutrons in operating PWRs for Type 316 stainless steel.

Test Temperature (°C)	Fluence x 10^{21} (n/cm^2)	Yield Stress (MPa)	Ultimate Stress (MPa)	Uniform Elongation (%)	Total Elongation (%)
23	75	938	1151	6	13
300	75	964	1031	0.8	7

The data, as compared to the unirradiated material properties, indicate an increase in yield stress and ultimate tensile stress and a drop in uniform and total elongation. These data for materials from PWR operating neutron fields correlate well with BWR data. Available BWR experimental data suggest that the following properties result after prolonged exposure of austenitic stainless steels and nickel-base alloys to neutron irradiation:

Estimated Irradiated Property	Base Metal	Weld Metal
Yield Stress (MPa)	850	850
Ultimate Stress (MPa)	850	850
Uniform Elongation (%)	0.5	0.5
Total Elongation (%)	5	5
Fracture Toughness (MPa-$m^{0.5}$)	50	40
Tearing Modulus	10	8

The above data show that, in irradiated BWR materials, the yield strength increases by about a factor of 3 over its unirradiated value while the ultimate stress remains essentially unchanged. The uniform elongation and the total elongation drop substantially from their unirradiated values. Fracture toughness drops by about a factor of 3, and the tearing modulus drops from several hundred to a range from 5 to 10.

4.1.2 Determination Of Significance

No instances of internals degradation have been recorded which have been definitely attributed to irradiation embrittlement. While the significance of irradiation embrittlement to the low alloy ferritic steels of PWR vessels has been established [4-5], its significance to the austenitic stainless steels of PWR RPV internals is less. The internals components most susceptible to irradiation embrittlement are those which are nearest to the reactor core. These components will experience significant neutron irradiation exposure, while remotely located components will receive relatively little. Embrittlement is evidenced by increases in yield stress and drops in elongation and fracture toughness, as described above in Section 4.1.1. In some cases, the noted increases in yield strength may even enhance the performance of components whose designs permit a limited amount of local yielding in the unirradiated condition.

Therefore, irradiation embrittlement can occur in PWR RPV internals components fabricated from austenitic stainless steel and nickel-base alloys with expected neutron fluences in excess of 1×10^{20} n/cm^2 ($E > 1$ MeV). If the expected neutron fluence is less than the threshold value of 1×10^{20} n/cm^2 ($E > 1$ MeV), then the changes in mechanical properties due to neutron exposure are insignificant. The PWR RPV internals components for which irradiation embrittlement is potentially significant are given in Table 4.1 and evaluated further in Section 5. A license renewal applicant intending to take credit for these conclusions is responsible for the review/evaluation of plant-specific features, including appropriate CLB documents/information in order to assure that the above assumptions and criteria are met.

Table 4.1

IRRADIATION EMBRITTLEMENT SUMMARY

for each STRUCTURAL ASSEMBLY GROUPING

Key: Degradation not significant □
Degradation potentially significant ■

UPPER INTERNALS ASSEMBLY

Westinghouse

Upper Internals Assembly

1. Upper Support Plate □
2. RCCA Guide Tube Assemblies □
   - RCCA Guide Tubes □
   - RCCA Guide Tube Bolts □
   - RCCA Guide Tube Support Pins □
3. Upper Support Columns □
   - Upper Support Column Bolts □
4. Upper Core Plate □
   - Fuel Pins □

Combustion Engineering

Upper Guide Structure Assembly

1. Upper Guide Structure Support Plate □
2. CEA Shroud Assemblies □
   - CEA Shrouds □
   - CEA Shroud Bolts □
3. ----------
4. Fuel Alignment Plate □

Babcock & Wilcox

Plenum Assembly

1. Plenum Cover and Plenum Cylinder □
2. CRA Guide Tube Assemblies □
   - CRA Guide Tubes □
   - CRA Guide Tube Bolts □
3. ----------
4. Upper Grid Assembly □
   - Upper Grid Rib Section □
   - Upper Grid Assembly Bolts □
   - Fuel Guide Pads □

CORE SUPPORT ASSEMBLY

Westinghouse

Core Barrel and Baffle/Former Assembly

1. Core Barrel ■
   - Upper Core Barrel Flange □
   - Core Barrel Nozzles □
2. ----------
3. Baffle-Former Assembly ■
   - Baffle / Former Assembly Bolts □
   - Baffle / Former Assembly Baffles □
4. Upper Core Plate Alignment Pins ■

Combustion Engineering

Core Support Barrel and Core Shroud Assembly

1. Core Support Barrel ■
   - Core Support Barrel Upper Flange □
2. ----------
3. Core Shroud Assembly ■
   - Core Shroud Assembly Bolts □
   - Core Shroud Tie Rods □
4. Fuel Alignment Plate Guide Lugs ■

Babcock & Wilcox

Core Support Assembly

1. Core Support Shield ■
   - Core Support Shield Flange □
2. Vent Valve Assemblies ■
3. Core Barrel Assembly ■
   - Core Barrel Bolts □
   - Baffle / Former Plates □
   - Baffle / Former Bolts □
4. ----------

LOWER INTERNALS ASSEMBLY

Westinghouse

Lower Internals Assembly

1. Lower Core Plate ■
2. Fuel Pins ■
3. Lower Support Plate ■
4. Lower Support Columns ■
   - Lower Support Column Bolts □
5. Radial Keys and Clevis Inserts □

Combustion Engineering

Core Support Assembly

1. Core Support Plate ■
2. Fuel Alignment Pins ■
3. Lower Support Structure Beam Assemblies ■
4. Core Support Columns ■
   - Core Support Column Bolts □
5. Core Support Barrel Snubber Assemblies ■

Babcock & Wilcox

Lower Grid Assembly

1. Lower Grid Top Rib Section ■
2. Fuel Guide Pads ■
3. Lower Grid Bottom Rib Weldment ■
4. Lower Grid Assembly Support Posts ■
   - Lower Grid Assembly Bolts □
5. Lower Grid Cylinder Guide Blocks ■

4.2 STRESS CORROSION CRACKING

4.2.1 Mechanism Description

Stress corrosion cracking (SCC) is localized non-ductile failure caused by a particular combination of stress, material condition, and environment. SCC can be intergranular or transgranular. It is generally accepted that tensile stresses greater than or equal to the yield strength of the material are required to induce SCC in a PWR environment. These stresses are the result of applied and/or residual loads such as stresses from cold working, welding, and thermal fluctuations. The material susceptibility varies from alloy to alloy and with the metallurgical condition of the material. For example, Type 304 stainless steel, a common PWR reactor internals component material of construction, may be more susceptible to SCC when it is in the sensitized condition and in an oxygenated environment. Environmental effects can also play a large part in causing SCC. The most aggressive contributors to SCC in a PWR environment are the oxygen concentration and the halogen levels.

4.2.2 Determination Of Significance

Initiation and propagation of SCC requires three factors to be present: (1) a susceptible material, (2) a corrosive environment and (3) the presence of tensile stresses. Two types of reactor internals components which have experienced SCC degradation are Alloy A-286 fasteners and Alloy X-750 support pins. Additional history on these fasteners is provided in Section 3.3.

B&W and CE reactor internals employ Alloy A-286 fasteners. A number of SCC failures of Alloy A-286 fasteners in B&W internals have occurred, prompting both vendors to perform evaluations [4-9 and 4-10]. The investigations performed by B&W concluded that the Alloy A-286 SCC susceptibility is increased with increasing stress and is dependent on the material condition (i.e., the fabrication process). The conclusions were that selecting fastener materials with design specification peak stress values < 100 ksi would preclude SCC [4-9]. These data generated by B&W were confirmed by operating experience in CE plants, where Alloy A-286 fasteners are used at significantly lower stress with no reported failures [4-10]. Therefore, the issues arising from stress corrosion cracking of these fasteners have been addressed by a variety of plant-specific approaches. Bolts which were inspected and found to be degraded have been replaced with bolts made of alternate designs of the same Alloy A-286 material in which the applied stresses were reduced or have been replaced with bolts made of an alternate high strength material (Alloy X-750). Bolts found to be in a nondegraded condition have been allowed to remain in place [4-9].

Westinghouse plants have had SCC failures of Alloy X-750 support pins in a number of operating PWRs [4-11]. The SCC observed in the Alloy X-750 pins used in Westinghouse RCCA guide tubes has been attributed to a specific heat treatment and high stress concentrations [4-11 and 4-12]. Replacement Alloy X-750 support pins have been fabricated with a new design to reduce stress and an improved heat treat condition which minimizes susceptibility to SCC in PWR environments.

A potential SCC concern also exists for other internals components. Some PWR RPV internal structures constructed of austenitic stainless steels have been subjected to low temperature heat treatments in the 900°F range during fabrication which may lead to sensitization, i.e., a metallurgical condition potentially susceptible to SCC. Other PWR internal components which may have a sensitized structure are the heat affected zones of welds (except possibly for those fabricated using low carbon weld rods). In these cases, along with susceptible material conditions, tensile stresses and aggressive environment must be present to promote SCC.

Regulatory Guide 1.44 [4-13], "Control of the Use of Sensitized Stainless Steel," was issued in 1973 and provides requirements to avoid sensitization of stainless steel that could lead to stress corrosion cracking. The requirements apply to austenitic stainless steels. PWR RPV internals components fabricated from austenitic stainless steel materials in the form of castings or weld metal with a ferrite content of at least 5 percent (corresponding to a Ferrite Number (FN) of 5 or greater) are exempted because they have not demonstrated susceptibility to stress corrosion cracking in a primary coolant environment.

Under normal PWR operating conditions, the oxygen concentration is monitored and controlled. During operation, when the temperatures make the material most susceptible to SCC, oxygen concentration is controlled to <5 ppb by externally-supplied hydrogen which leads to excess dissolved hydrogen in the primary coolant water. During plant cooldown when oxygen injection is used to control CRUD-bursts in order to clean up the primary coolant in preparation for shutdown, the control of oxygen levels versus temperature will govern cooldown rates in order to avoid conditions conducive to SCC. At shutdown conditions, when the primary coolant may be exposed to air, the oxygen levels are monitored and controlled. Primary water chemistry controls remain in place during shutdown so even though the fluid temperatures (i.e. thermal stresses) are low during shutdown, these controls will mitigate the potential for SCC of reactor internals components. This continuous control of oxygen eliminates one of the most aggressive environmental factors that could lead to SCC.

Furthermore, in addition to the control of radiolytic dissolved oxygen by excess dissolved hydrogen, stagnant (crevice) regions are protected by [4-14]: (1) diffusion of hydrogen from the bulk coolant, (2) enhanced recombination rates from the effects of the crevice on the radiation quality, and (3) rapid reaction of the oxygen with the materials of construction leaving an excess of hydrogen in the stagnant areas. Also, the levels of halogens (i.e., chlorides and fluorides) are maintained below 150 ppb during all phases of operation. This low halogen level will prevent the occurrence of SCC of austenitic stainless steel as shown by Figure 4.1 [4-15] for chlorides. BWR results [4-47] have also shown the importance of low oxygen and low halogen levels.

Therefore, for reactor internals components fabricated from austenitic stainless steel, provided that the components are subjected to stress levels within the material design specification, and the PWR operating chemistry can be shown to limit the oxygen to < 5 ppb and the halogens to < 150 ppb, then SCC is not a significant age-related degradation mechanism. Otherwise, further evaluation is required and is provided in Section 5.

For reactor internals components fabricated from nickel-base alloys, SCC is a potentially significant age-related degradation mechanism. Further evaluation of these components is required and is provided in Section 5.

The fasteners identified in Table 4.2 potentially fall into these two groupings. For all other components, SCC is considered nonsignificant and requires no further evaluation. Procedures for evaluating the age-related degradation effects of SCC on the fasteners identified are provided in Section 5. A license renewal applicant intending to take credit for these conclusions is responsible for the review/evaluation of plant-specific features, including appropriate CLB documents/information in order to assure that the above assumptions and criteria are met.

FIGURE 4.1
THE EFFECTS OF OXYGEN AND CHLORIDE ON THE SCC OF AUSTENITIC STAINLESS STEEL IN HIGH-TEMPERATURE WATER [4-15]

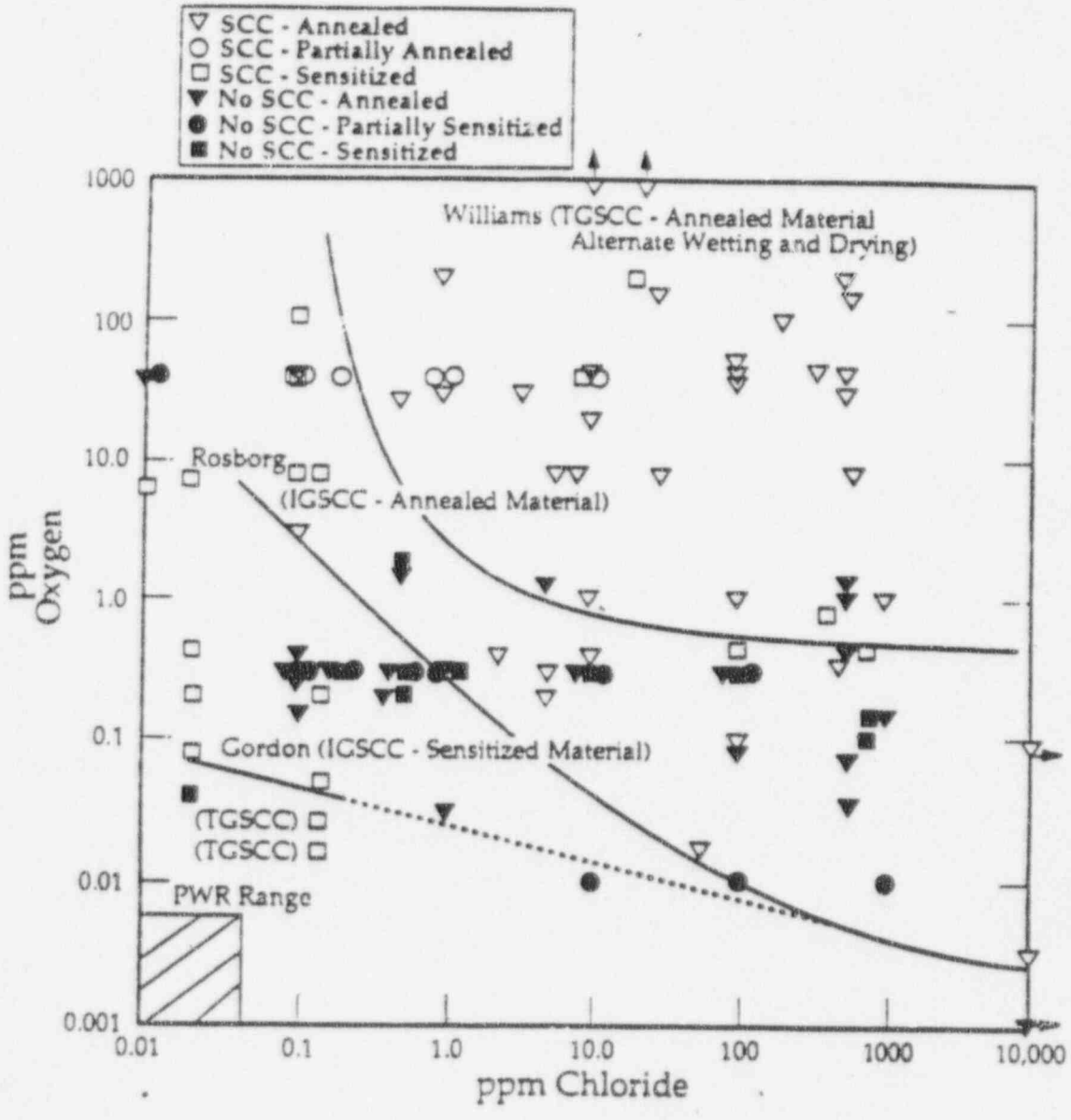


Table 4.2

STRESS CORROSION CRACKING SUMMARY

for each STRUCTURAL ASSEMBLY GROUPING

Key: Degradation not significant ☐
Degradation potentially significant ■

UPPER INTERNALS ASSEMBLY

Westinghouse		Combustion Engineering		Babcock & Wilcox	
Upper Internals Assembly		**Upper Guide Structure Assembly**		**Plenum Assembly**	
1. Upper Support Plate	☐	1. Upper Guide Structure Support Plate	☐	1. Plenum Cover and Plenum Cylinder	☐
2. RCCA Guide Tube Assemblies	☐	2. CEA Shroud Assemblies	☐	2. CRA Guide Tube Assemblies	☐
RCCA Guide Tubes	☐	CEA Shrouds	☐	CRA Guide Tubes	☐
RCCA Guide Tube Bolts	☐	CEA Shroud Bolts	■	CRA Guide Tube Bolts	☐
RCCA Guide Tube Support Pins	■				
3. Upper Support Columns	☐	3. ----------		3. ----------	
Upper Support Column Bolts	☐				
4. Upper Core Plate	☐	4. Fuel Alignment Plate	☐	4. Upper Grid Assembly	☐
Fuel Pins	☐			Upper Grid Rib Section	☐
				Upper Grid Assembly Bolts	☐
				Fuel Guide Pads	☐

CORE SUPPORT ASSEMBLY

Core Barrel and Baffle/Former Assembly		**Core Support Barrel and Core Shroud Assembly**		**Core Support Assembly**	
1. Core Barrel	☐	1. Core Support Barrel	☐	1. Core Support Shield	☐
Upper Core Barrel Flange	☐	Core Support Barrel Upper Flange	☐	Core Support Shield Flange	☐
Core Barrel Nozzles	☐				
2. ----------		2. ----------		2. Vent Valve Assemblies	☐
3. Baffle-Former Assembly	☐	3. Core Shroud Assembly	☐	3. Core Barrel Assembly	☐
Baffle / Former Assembly Bolts	☐	Core Shroud Assembly Bolts	☐	Core Barrel Bolts	■
Baffle / Former Assembly Baffles	☐	Core Shroud Tie Rods	☐	Baffle / Former Plates	☐
				Baffle / Former Bolts	☐
4. Upper Core Plate Alignment Pins	☐	4. Fuel Alignment Plate Guide Lugs	☐	4. ----------	

LOWER INTERNALS ASSEMBLY

Lower Internals Assembly		**Core Support Assembly**		**Lower Grid Assembly**	
1. Lower Core Plate	☐	1. Core Support Plate	☐	1. Lower Grid Top Rib Section	☐
2. Fuel Pins	☐	2. Fuel Alignment Pins	☐	2. Fuel Guide Pads	☐
3. Lower Support Plate	☐	3. Lower Support Structure Beam Assemblies	☐	3. Lower Grid Bottom Rib Weldment	☐
4. Lower Support Columns	☐	4. Core Support Columns	☐	4. Lower Grid Assembly Support Posts	☐
Lower Support Column Bolts	☐	Core Support Column Bolts	☐	Lower Grid Assembly Bolts	☐
5. Radial Keys and Clevis Inserts	☐	5. Core Support Barrel Snubber Assemblies	☐	5. Lower Grid Cylinder Guide Blocks	☐

4.3 IRRADIATION-ASSISTED STRESS CORROSION CRACKING

4.3.1 Mechanism Description

Experience with boiling water reactors (BWRs) suggests that irradiation-assisted stress corrosion cracking (IASCC) is a plausible aging mechanism for the later stages of PWR internals life. Two primary mechanisms can lead to accelerated irradiation-assisted stress corrosion cracking in a reactor environment. First, the microstructural changes resulting from the combination of long-term thermal aging and radiation damage in the material may reduce the resistance to stress corrosion cracking. Radiation-enhanced segregation of solute species to grain boundaries may lead to a pseudo-sensitization of the material. Radiation may also lead to enhanced crack tip strain rates that could promote stress corrosion cracking. An evaluation of the potential of these radiation damage mechanisms [4-16] concludes that radiation-enhanced segregation is the dominant mechanism for inducing stress corrosion cracking. The second mechanism for increasing the rate of stress corrosion cracking involves alterations in the local water chemistry caused by irradiation.

4.3.2 Determination Of Significance

IASCC may be a potentially significant age-related degradation mechanism for PWR RPV internals components, provided that the fluence level is greater than 5×10^{20} n/cm² for materials with low stress levels (< 10 ksi) and 1×10^{20} n/cm² for moderate to high stress levels (> 10 ksi). Under normal PWR operating conditions, the oxygen concentration (and, thus, the electrochemical potential) is controlled by the addition of hydrogen sufficient to ensure excess dissolved hydrogen [4-14]. In this way, any dissolved oxygen derived from radiolytic decomposition is limited to < 5 ppb. Furthermore, the levels of halogens (i.e., chlorides and fluorides) are maintained below 150 ppb during all phases of operation. For the BWR coolant environment, reference [4-46] states that "the same environmental solution for intergranular SCC of piping (hydrogen water chemistry) appears promising for this phenomenon" (IASCC). Therefore, even though many PWR reactor internals components have received exposures beyond the neutron fluence threshold of 5×10^{20} n/cm² (E > 1 MeV) established for BWR internals components [4-17], the relatively benign environment of the PWR coolant, which incorporates hydrogen overpressure and achieves reduced oxygen levels in comparison to BWR coolant, reduces (but does not necessarily eliminate) the potential for IASCC degradation of PWR reactor internals components.

Therefore, provided that the PWR primary coolant is maintained such that oxygen concentrations are < 5 ppb and halogen concentrations are < 150 ppb, the fluence level is less than 5×10^{20} n/cm² for materials with low stress levels (1×10^{20} n/cm² for moderate to high stress levels) and the operating stresses are within the design envelope, IASCC is not a significant age-related degradation mechanism for PWR RPV internals components. For RPV internals components that are not enveloped by the above conditions, IASCC is a potentially significant age-related degradation mechanism. If neutron flux maps or measurements are available that show the fluence level to be below the threshold levels of concern (5×10^{20} or 1×10^{20} n/cm²) for certain components, then those components will not be susceptible to

IASCC. However, for the purposes of this report, it will be assumed that fluence levels exceed these threshold values and all PWR RPV internals components will be considered potentially susceptible to IASCC. Further evaluation is required and is provided in Section 5.

A license renewal applicant intending to take credit for these conclusions is responsible for the review/evaluation of plant-specific features, including appropriate CLB documents/information in order to assure that the above assumptions and criteria are met.

Table 4.3

IRRADIATION-ASSISTED STRESS CORROSION CRACKING SUMMARY

for each STRUCTURAL ASSEMBLY GROUPING

Key: ☐ Degradation not significant
■ Degradation potentially significant

UPPER INTERNALS ASSEMBLY

Westinghouse		Combustion Engineering		Babcock & Wilcox	
Upper Internals Assembly		**Upper Guide Structure Assembly**		**Plenum Assembly**	
1. Upper Support Plate	■	1. Upper Guide Structure Support Plate	■	1. Plenum Cover and Plenum Cylinder	☐
2. RCCA Guide Tube Assemblies	■	2. CEA Shroud Assemblies	■	2. CRA Guide Tube Assemblies	■
RCCA Guide Tubes	☐	CEA Shrouds	☐	CRA Guide Tubes	☐
RCCA Guide Tube Bolts	☐	CEA Shroud Bolts	☐	CRA Guide Tube Bolts	☐
RCCA Guide Tube Support Pins	☐				
3. Upper Support Columns	■	3. ----------		3. ----------	
Upper Support Column Bolts	☐				
4. Upper Core Plate	■	4. Fuel Alignment Plate	■	4. Upper Grid Assembly	■
Fuel Pins	☐			Upper Grid Rib Section	☐
				Upper Grid Assembly Bolts	☐
				Fuel Guide Pads	☐

CORE SUPPORT ASSEMBLY

Core Barrel and Baffle/Former Assembly		**Core Support Barrel and Core Shroud Assembly**		**Core Support Assembly**	
1. Core Barrel	■	1. Core Support Barrel	■	1. Core Support Shield	■
Upper Core Barrel Flange	☐	Core Support Barrel Upper Flange	☐	Core Support Shield Flange	☐
Core Barrel Nozzles	☐				
2. ----------		2. ----------		2. Vent Valve Assemblies	■
3. Baffle-Former Assembly	■	3. Core Shroud Assembly	■	3. Core Barrel Assembly	■
Baffle / Former Assembly Bolts	☐	Core Shroud Assembly Bolts	☐	Core Barrel Bolts	☐
Baffle / Former Assembly Baffles	☐	Core Shroud Tie Rods	☐	Baffle / Former Plates	☐
				Baffle / Former Bolts	☐
4. Upper Core Plate Alignment Pins	■	4. Fuel Alignment Plate Guide Lugs	■	4. ----------	

LOWER INTERNALS ASSEMBLY

Lower Internals Assembly		**Core Support Assembly**		**Lower Grid Assembly**	
1. Lower Core Plate	■	1. Core Support Plate	■	1. Lower Grid Top Rib Section	■
2. Fuel Pins	■	2. Fuel Alignment Pins	■	2. Fuel Guide Pads	■
3. Lower Support Plate	■	3. Lower Support Structure Beam Assemblies	■	3. Lower Grid Bottom Rib Weldment	■
4. Lower Support Columns	■	4. Core Support Columns	■	4. Lower Grid Assembly Support Posts	■
Lower Support Column Bolts	☐	Core Support Column Bolts	☐	Lower Grid Assembly Bolts	☐
5. Radial Keys and Clevis Inserts	■	5. Core Support Barrel Snubber Assemblies	■	5. Lower Grid Cylinder Guide Blocks	■

4.4 EROSION AND EROSION/CORROSION

4.4.1 Mechanism Description

Erosion is attributed to the removal of protective surface films on a metal by mechanical action of a fluid or particulate matter. Erosion/corrosion occurs when the fluid or particulate matter is also corrosive to the metal. General erosion occurs under high-velocity conditions, turbulence, and impingement. Geometrical factors are extremely important. Carbon steels and low alloy steels are most susceptible to erosion/corrosion. Higher alloy steels, nickel-base alloys, and stainless steels are considered resistant to erosion and erosion/corrosion in a PWR environment.

4.4.2 Determination Of Significance

All of the PWR reactor internals components being considered here are constructed of austenitic stainless steel or nickel-base alloys that are considered to be resistant to erosion and erosion/corrosion in a PWR environment. The relatively low fluid flow velocities in the PWR RPV do not permit erosion to become a significant age-related degradation issue. The pH levels in the bulk coolant minimize corrosion and thus erosion/corrosion. The operating pressures of a PWR (approximately 2250 psia) preclude cavitation erosion, and the purity and particulate control of the reactor coolant eliminate particulate erosion as a significant concern [4-18]. Therefore, erosion and erosion/corrosion are not significant age-related degradation mechanisms for any of the PWR reactor internals components considered in this report. A license renewal applicant intending to take credit for these conclusions is responsible for the review/evaluation of plant-specific features, including appropriate CLB documents/information in order to assure that the above assumptions and criteria are met.

4.5 CREEP

4.5.1 Mechanism Description

Creep is the plastic deformation which occurs over a period of time in a material subjected to a stress which is typically below the elastic limit. Creep occurs at elevated temperatures where continuous deformation takes place under constant strain. Creep is not a concern for austenitic alloys below 1000°F and nickel-base alloys below 1400°F [4-19].

4.5.2 Determination Of Significance

The maximum temperature experienced by PWR RPV internals during normal and upset conditions is approximately 650°F, except for certain localized areas where temperatures can be as high as 800°F for short durations. Even with localized temperature excursions to 800°F, these temperatures are well below the temperatures at which creep is a concern for any of the austenitic stainless steel and nickel-base alloy PWR RPV internals components (note specific material types in Table 3.3). Therefore, creep is not a significant age-related degradation mechanism for any PWR RPV internals component. A license renewal applicant intending to take credit for this conclusion is responsible for the review/evaluation of plant-specific features, including appropriate CLB documents/information in order to assure that the above assumptions and criteria are met.

4.6 STRESS RELAXATION

4.6.1 Mechanism Description

Stress relaxation is the unloading of pre-loaded components caused by long term exposure of internals materials to elevated temperatures and/or neutron irradiation. It occurs under conditions of constant strain where part of the elastic strain is replaced with plastic strain. At temperatures well above operating temperatures, the thermal effect is predominant. It has been determined, however, that the presence of fast neutron irradiation can result in stress relaxation even at normal operating temperatures [4-20 and 4-21]. When the irradiation effect is dominant, the rate of neutron impingement controls the number of vacancies formed in the component material. The presence of vacancies increases the likelihood that the material will plastically deform, resulting in the relaxation effect. Stress relaxation is particularly important in the design of bolted connections.

4.6.2 Determination Of Significance

Stress relaxation has significance only to components with substantial preloads, such as torqued bolts, because the maintenance of an adequate preload is important to their functionality. A loss of preload in such components could result in higher cyclic and transient loads and an increase in fatigue susceptibility (Section 4.10). In some reactor internals components, stress relaxation is accounted for in the design by preload margins that are incorporated to compensate for the effects of thermal transients. Neutron irradiation can, over an extended period of time and depending on the fluence, lead to stress relaxation of preloaded components.

The influence of irradiation and temperature on stress relaxation behavior has been evaluated for materials stressed at or above the yield strength and exposed to irradiation [4-22 through 4-27]. Data have shown that the thermal effects on annealed Type 304 stainless steel at temperatures below 900°F produce a stress relaxation maximum of about 18% [4-23]. The combined effects of temperature and irradiation have been reported to produce further relaxation. In one report, significant stress relaxation, as high as 60 to 100%, occurred in annealed Type 304, Alloy X-750 and Alloy 718 during irradiation to fluences equal to or greater than 5×10^{20} (E > 1 MeV) [4-24]. Similar data reported that Type 304 bolts relaxed from a 20 ksi cold preload stress to 10 ksi after reactor heatup to 550°F and irradiation at 6×10^{19} n/cm^2 (E > 1 MeV) [4-25]. Two additional evaluations have shown a relaxation of 35% in annealed Type 304 stainless steel after irradiation at 4×10^{20} n/cm^2 (E > 1 MeV) [4-26 and 4-27]. These data suggest that the combined effects of temperature and irradiation can potentially result in significant stress relaxation of preloaded components.

Therefore, provided that the internals components do not depend on preload for functionality (e.g., bolting), stress relaxation is not a significant age-related degradation mechanism affecting the license renewal of PWR reactor internals components. Otherwise, stress relaxation is potentially significant and requires further evaluation as provided in Section 5. The PWR internals components for which stress relaxation is potentially significant are given

in Table 4.4. Note that stress relaxation is potentially significant only for fasteners which require preload in order to maintain functionality. A license renewal applicant intending to take credit for this conclusion is responsible for the review/evaluation of plant-specific features, including appropriate CLB documents/information in order to assure that the above assumptions and criteria are met.

Table 4.4

STRESS RELAXATION SUMMARY

for each STRUCTURAL ASSEMBLY GROUPING

Key: Degradation not significant ☐
Degradation potentially significant ■

Westinghouse		Combustion Engineering		Babcock & Wilcox	
UPPER INTERNALS ASSEMBLY					
Upper Internals Assembly		**Upper Guide Structure Assembly**		**Plenum Assembly**	
1. Upper Support Plate	☐	1. Upper Guide Structure Support Plate	☐	1. Plenum Cover and Plenum Cylinder	☐
2. RCCA Guide Tube Assemblies	☐	2. CEA Shroud Assemblies	☐	2. CRA Guide Tube Assemblies	☐
RCCA Guide Tubes	☐	CEA Shrouds	☐	CRA Guide Tubes	☐
RCCA Guide Tube Bolts	■	CEA Shroud Bolts	■	CRA Guide Tube Bolts	■
RCCA Guide Tube Support Pins	☐				
3. Upper Support Columns	☐	3. ----------		3. ----------	
Upper Support Column Bolts	■				
4. Upper Core Plate	☐	4. Fuel Alignment Plate	☐	4. Upper Grid Assembly	☐
Fuel Pins	☐			Upper Grid Rib Section	☐
				Upper Grid Assembly Bolts	■
				Fuel Guide Pads	☐
CORE SUPPORT ASSEMBLY					
Core Barrel and Baffle/Former Assembly		**Core Support Barrel and Core Shroud Assembly**		**Core Support Assembly**	
1. Core Barrel	☐	1. Core Support Barrel	☐	1. Core Support Shield	☐
Upper Core Barrel Flange	☐	Core Support Barrel Upper Flange	☐	Core Support Shield Flange	☐
Core Barrel Nozzles	☐				
2. ----------		2. ----------		2. Vent Valve Assemblies	☐
3. Baffle-Former Assembly	☐	3. Core Shroud Assembly	☐	3. Core Barrel Assembly	☐
Baffle / Former Assembly Bolts	■	Core Shroud Assembly Bolts	■	Core Barrel Bolts	■
Baffle / Former Assembly Baffles	☐	Core Shroud Tie Rods	■	Baffle / Former Plates	☐
				Baffle / Former Bolts	■
4. Upper Core Plate Alignment Pins	☐	4. Fuel Alignment Plate Guide Lugs	☐	4. ----------	
LOWER INTERNALS ASSEMBLY					
Lower Internals Assembly		**Core Support Assembly**		**Lower Grid Assembly**	
1. Lower Core Plate	☐	1. Core Support Plate	☐	1. Lower Grid Top Rib Section	☐
2. Fuel Pins	■	2. Fuel Alignment Pins	■	2. Fuel Guide Pads	☐
3. Lower Support Plate	☐	3. Lower Support Structure Beam Assemblies	☐	3. Lower Grid Bottom Rib Weldment	☐
4. Lower Support Columns	☐	4. Core Support Columns	☐	4. Lower Grid Assembly Support Posts	☐
Lower Support Column Bolts	■	Core Support Column Bolts	■	Lower Grid Assembly Bolts	■
5. Radial Keys and Clevis Inserts	☐	5. Core Support Barrel Snubber Assemblies	☐	5. Lower Grid Cylinder Guide Blocks	☐

4.7 WEAR

4.7.1 Mechanism Description

Wear is defined as the removal of material surface layers due to relative motion between two surfaces or under the influence of hard, abrasive particles. Wear occurs in parts that experience intermittent relative motion, in clamped joints where relative motion is not intended but may occur due to a loss of clamping force or as a result of flow-induced vibrations.

Wear, which is the result of the contact of two surfaces due to vibration or sliding (e.g., flow-induced vibration) while the surfaces are in the presence of a corrosive environment, is referred to as fretting wear. Another type of wear which may occur in PWRs and which is not related to flow-induced vibration is that associated with the intentional displacement of adjacent components, such as the stepping of control rods. Wear can result from either surface oxide removal or the direct removal of base material.

4.7.2 Determination Of Significance

Wear has been observed at the interfaces of components whose relative motion is not completely restrained. In the reactor internals components, wear between control rods and guide tubes results from the axial sliding which occurs during insertions and withdrawals, and also from the transverse motions caused by flow-induced vibration. Measurements of control rod wear infer that wear on guide tube inner surfaces has also occurred.

Other components considered to be potentially susceptible to wear degradation are those which constitute the interfaces between structural components. The core barrel flange, radial keys/clevises, fuel pins and upper core plate alignment pins are examples of such interfacing components in Westinghouse plants. Similar interfacing components exist in Combustion Engineering and Babcock and Wilcox plants. The predominant excitation mechanism for wear of these components is flow-induced vibration, although thermal effects can also contribute by increasing interface loads and by directly producing relative thermal displacements between interfacing components.

Wear is a potentially significant age-related degradation mechanism for the internals components identified in Table 4.5. Further evaluation of wear with respect to these components is provided in Section 5. For all other PWR reactor internals components covered by this report, wear is nonsignificant. A license renewal applicant intending to take credit for this conclusion is responsible for the review/evaluation of plant-specific features, including appropriate CLB documents/information in order to assure that the above assumptions and criteria are met.

Table 4.5

WEAR SUMMARY

for each STRUCTURAL ASSEMBLY GROUPING

Key: Degradation not significant □
Degradation potentially significant ■

UPPER INTERNALS ASSEMBLY

Westinghouse		Combustion Engineering		Babcock & Wilcox	
Upper Internals Assembly		**Upper Guide Structure Assembly**		**Plenum Assembly**	
1. Upper Support Plate	□	1. Upper Guide Structure Support Plate	□	1. Plenum Cover and Plenum Cylinder	□
2. RCCA Guide Tube Assemblies	□	2. CEA Shroud Assemblies	□	2. CRA Guide Tube Assemblies	□
RCCA Guide Tubes	■	CEA Shrouds	■	CRA Guide Tubes	■
RCCA Guide Tube Bolts	□	CEA Shroud Bolts	□	CRA Guide Tube Bolts	□
RCCA Guide Tube Support Pins	□				
3. Upper Support Columns	□	3. ----------		3. ----------	
Upper Support Column Bolts	□				
4. Upper Core Plate	□	4. Fuel Alignment Plate	■	4. Upper Grid Assembly	□
Fuel Pins	■			Upper Grid Rib Section	□
				Upper Grid Assembly Bolts	□
				Fuel Guide Pads	■

CORE SUPPORT ASSEMBLY

Core Barrel and Baffle/Former Assembly		**Core Support Barrel and Core Shroud Assembly**		**Core Support Assembly**	
1. Core Barrel	□	1. Core Support Barrel	□	1. Core Support Shield	□
Upper Core Barrel Flange	■	Core Support Barrel Upper Flange	■	Core Support Shield Flange	■
Core Barrel Nozzles	□				
2. ----------		2. ----------		2. Vent Valve Assemblies	□
3. Baffle-Former Assembly	□	3. Core Shroud Assembly	□	3. Core Barrel Assembly	□
Baffle / Former Assembly Bolts	□	Core Shroud Assembly Bolts	□	Core Barrel Bolts	□
Baffle / Former Assembly Baffles	□	Core Shroud Tie Rods	□	Baffle / Former Plates	□
				Baffle / Former Bolts	□
4. Upper Core Plate Alignment Pins	■	4. Fuel Alignment Plate Guide Lugs	■	4. ----------	

LOWER INTERNALS ASSEMBLY

Lower Internals Assembly		**Core Support Assembly**		**Lower Grid Assembly**	
1. Lower Core Plate	□	1. Core Support Plate	□	1. Lower Grid Top Rib Section	□
2. Fuel Pins	■	2. Fuel Alignment Pins	■	2. Fuel Guide Pads	■
3. Lower Support Plate	□	3. Lower Support Structure Beam Assemblies	□	3. Lower Grid Bottom Rib Weldment	□
4. Lower Support Columns	□	4. Core Support Columns	□	4. Lower Grid Assembly Support Posts	□
Lower Support Column Bolts	□	Core Support Column Bolts	□	Lower Grid Assembly Bolts	□
5. Radial Keys and Clevis Inserts	■	5. Core Support Barrel Snubber Assemblies	■	5. Lower Grid Cylinder Guide Blocks	■

4.8 THERMAL AGING

4.8.1 Mechanism Description

The ferritic phase of cast austenitic stainless steels (CASS) increases the tensile strength and serves to improve the weldability, stress corrosion resistance, and casting soundness. Inservice thermal exposure of the delta ferrite results in further increases in strength and a decrease in the ductility and fracture toughness properties of the material. Operating temperatures for the reactor internals components generally range between 550 and 650°F. At these temperatures, cast austenitic materials exhibit a spinodal decomposition of the ferrite phase into chromium-rich and ferrite-rich phases. Although initially very tough (i.e., exhibiting high fracture toughness), these complex metallurgical changes may result in potentially significant embrittlement (i.e., reduction in fracture toughness) depending on the amount, morphology, and distribution of the ferrite as well as on the precipitation reactions that occur in the ferrite. The effect of the embrittlement on toughness is usually measured in terms of Charpy impact energy, elastic-plastic fracture toughness (J_{Ic}), and/or the tearing modulus. For long-term aging, such as 40 years at operating temperatures, reductions in fracture toughness approaching an order of magnitude are indicated based on current information. Even so, the cast austenitic stainless steel still exhibits ductile behavior and resistance to crack extension.

The CASS thermal embrittlement phenomenon is both time and temperature dependent. The temperature dependence of the rate of embrittlement can be characterized in terms of activation energy principles, with the maximum rate of embrittlement occurring in the temperature range between 840 and 930°F; hence, the commonly-used name of "885°F embrittlement." At lower temperatures the embrittlement rate is less, but the effects of thermal embrittlement have been observed at temperatures as low as 550 to 650°F, the operating range for the reactor internals components. In order to accelerate the rate of embrittlement, most of the experimental studies of this mechanism have been carried out at temperatures above 690°F, with extrapolation of the results to lower temperatures (and, correspondingly, to lower rates of embrittlement and longer aging time) through Arrhenius models [4-28 through 4-30].

Alloy composition also has an effect. The presence of higher molybdenum (Mo) content in CF3M and CF8M grades has been shown to have an influence on the aging process [4-31]. The greater loss of toughness in the higher carbon grades (CF8 and CF8M) can be attributed to the weakening of the austenite-ferrite boundaries caused by carbide precipitation. Other alloy constituents that influence the thermal aging process are the ferrite formers, such as nitrogen. Equally important is the casting process used to fabricate the component. Centrifugally-cast components are more resistant to thermal aging than statically-cast components [4-33].

Thermal aging can also have an effect on martensitic stainless steel as it exists in its time- and temperature-dependent, metastable state. Very few reactor internals components are manufactured from martensitic stainless steel.

4.8.2 Determination Of Significance

The cast austenitic stainless steel PWR reactor internals components are exposed to temperatures that could lead to eventual thermal aging embrittlement, provided that the term of exposure is sufficiently long and that the other factors that control the extent of embrittlement (e.g., casting process, delta ferrite, material chemistry) are unfavorable. The PWR RPV internals components fabricated from CASS are identified in Table 4.6.

For centrifugally-cast component parts with delta ferrite content below 20%, mechanical properties are not degraded significantly by the thermal aging embrittlement process [4-32 and 4-33]. For statically-cast component parts with Mo content such that it meets casting grade CF3 or CF8 limits, the 20% delta ferrite threshold also applies. However, for statically-cast component parts with Mo content above that meeting CF3 or CF8 limits, 14% delta ferrite is the threshold below which no significant degradation due to thermal aging is observed [4-32].

In summary, thermal aging embrittlement is a potentially significant age-related degradation mechanism for the CASS PWR RPV internals components exceeding the following threshold criteria and further evaluation is provided in Section 5.

Thermal aging embrittlement is potentially significant for :

(1) Centrifugally-cast component parts, with a delta ferrite content above 20%.
(2) Statically-cast component parts, with Mo content meeting CF3 and CF8 limits and with a delta ferrite content above 20%.
(3) Statically-cast component parts, with Mo content exceeding CF3 and CF8 limits with a delta ferrite content above 14%.

Thermal aging embrittlement is also a potentially significant age-related degradation mechanism for the martensitic stainless steel subcomponents in the B&W vent valve assemblies. Further evaluation of these vent valve assemblies is provided in Section 5.

A license renewal applicant intending to take credit for these conclusions is responsible for the review/evaluation of plant-specific features, including appropriate CLB documents/information in order to assure that the above assumptions and criteria are met.

Table 4.6

THERMAL AGING SUMMARY

for each STRUCTURAL ASSEMBLY GROUPING

Key: ☐ Degradation not significant
■ Degradation potentially significant

UPPER INTERNALS ASSEMBLY

Westinghouse		Combustion Engineering		Babcock & Wilcox	
Upper Internals Assembly		**Upper Guide Structure Assembly**		**Plenum Assembly**	
1. Upper Support Plate	☐	1. Upper Guide Structure Support Plate	☐	1. Plenum Cover and Plenum Cylinder	☐
2. RCCA Guide Tube Assemblies	☐	2. CEA Shroud Assemblies	☐	2. CRA Guide Tube Assemblies	☐
RCCA Guide Tubes	☐	CEA Shrouds *	■	CRA Guide Tubes *	■
RCCA Guide Tube Bolts	☐	CEA Shroud Bolts	☐	CRA Guide Tube Bolts	☐
RCCA Guide Tube Support Pins	☐				
3. Upper Support Columns *	■	3. ----------		3. ----------	
Upper Support Column Bolts	☐				
4. Upper Core Plate	☐	4. Fuel Alignment Plate	☐	4. Upper Grid Assembly	☐
Fuel Pins	☐			Upper Grid Rib Section	☐
				Upper Grid Assembly Bolts	☐
				Fuel Guide Pads	☐

CORE SUPPORT ASSEMBLY

Core Barrel and Baffle/Former Assembly		**Core Support Barrel and Core Shroud Assembly**		**Core Support Assembly**	
1. Core Barrel	☐	1. Core Support Barrel	☐	1. Core Support Shield	☐
Upper Core Barrel Flange	☐	Core Support Barrel Upper Flange	☐	Core Support Shield Flange	☐
Core Barrel Nozzles	☐				
2. ----------		2. ----------		2. Vent Valve Assemblies	■
3. Baffle-Former Assembly	☐	3. Core Shroud Assembly	☐	3. Core Barrel Assembly	☐
Baffle / Former Assembly Bolts	☐	Core Shroud Assembly Bolts	☐	Core Barrel Bolts	☐
Baffle / Former Assembly Baffles	☐	Core Shroud Tie Rods	☐	Baffle / Former Plates	☐
				Baffle / Former Bolts	☐
4. Upper Core Plate Alignment Pins	☐	4. Fuel Alignment Plate Guide Lugs	☐	4. ----------	

LOWER INTERNALS ASSEMBLY

Lower Internals Assembly		**Core Support Assembly**		**Lower Grid Assembly**	
1. Lower Core Plate	☐	1. Core Support Plate	☐	1. Lower Grid Top Rib Section	☐
2. Fuel Pins	☐	2. Fuel Alignment Pins	☐	2. Fuel Guide Pads	☐
3. Lower Support Plate *	■	3. Lower Support Structure Beam Assemblies	☐	3. Lower Grid Bottom Rib Weldment	☐
4. Lower Support Columns *	■	4. Core Support Columns *	■	4. Lower Grid Assembly Support Posts	☐
Lower Support Column Bolts	☐	Core Support Column Bolts	☐	Lower Grid Assembly Bolts	☐
5. Radial Keys and Clevis Inserts	☐	5. Core Support Barrel Snubber Assemblies	☐	5. Lower Grid Cylinder Guide Blocks	☐

* - ONLY THOSE PARTS OR SUBPARTS CONSTRUCTED OF CAST AUSTENITIC STAINLESS STEEL

4.9 CORROSION

4.9.1 Mechanism Description

Corrosion is an electrochemical reaction between a metal or alloy and its environment and is characterized by a deterioration of the materials. Two forms of corrosion are typically defined: (1) general corrosion, and (2) pitting or crevice corrosion. In both instances, the material becomes thinner by chemical attack (dissolution). General corrosion is characterized as the thinning or wastage of a material, more or less uniformly, by an aggressive environment. Pitting, or crevice, corrosion occurs when surfaces of materials are wetted by the corrosion medium and are covered in localized areas with debris (e.g., crud); used in contact with each other; or when a crack or crevice is permitted to exist in a component exposed to such media. The consequences of such damage are wall thinning, reduction of cross-sectional area, material removal, and build-up of corrosion products [4-34].

4.9.2 Determination Of Significance

As delineated in Table 3.3, the materials used in the construction of the reactor internals components are austenitic stainless steel and nickel-base alloys. These alloys are not susceptible to general corrosion in the benign PWR primary coolant because they passivate to form protective layers which mitigate the potential for corrosion degradation.

This resistance extends to crevice regions, such as bolt threads, where an aggressive environment has the potential to cause localized corrosion, even for these film-forming materials. The mechanism of crevice corrosion involves an electrochemical action between the surfaces of the crevice and those exposed freely to the environment outside it [4-35]. In the simplest case of an oxygenated environment, this would be a differential aeration or oxygen-concentration cell effect, with the metal within the crevice (which is shielded from free access to dissolved oxygen) becoming anodic to the outside surfaces freely exposed to the oxygen-bearing solution. With austenitic stainless steels or nickel-base alloys that resist corrosion as a result of a passive state, the matter becomes complicated and aggravated by the active-passive galvanic cells that may become established. The metal within the crevice that is shielded from oxygen may become active, while that outside in free contact with oxygen will maintain its passivity. The difference in potential between the active and passive states will become a driving force in the galvanic corrosion of the anode within the crevice. It can be assumed therefore that the crevice corrosion of a passive metal is simply a particular kind of galvanic cell, with the anode being the active metal within the crevice and the cathode the passive metal outside it. This being the case, the action of this cell should obey the rules of other galvanic cells. In particular, in environments where the galvanic cell reaction is under cathodic control (e.g., where the supply of oxygen to depolarize the cathodes has a controlling effect) the extent of corrosion of the anodes within the crevice should be determined principally by the area of the freely exposed surfaces (cathodes) outside the crevices.

Corrosion in crevices can occur because of [4-36] (1) an increase in metal-ion concentration within the crevice as compared with the concentration outside the crevice, which may then

result in concentration-cell corrosion; (2) a decrease in oxygen concentration inside the crevice, which may then result in oxygen-concentration-cell corrosion; or (3) increased corrosivity resulting from the accumulation of soluble and insoluble corrosion products formed within the crevice (this is also known as "stagnant area corrosion"). All three of these mechanisms are the result of little or no fluid circulation through the crevice. The following variables have been identified as important in the consideration of crevice corrosion: composition of the material, oxygen content, hydrogen content, pH, and temperature.

In many crevice tests conducted at elevated temperature with different materials, it was found that the extent of crevice corrosion generally decreased with the increasing corrosion resistance of the material tested [4-36]. Straight chromium stainless steels and nickel-alloys were subject to slight pitting in crevice areas, whereas austenitic stainless steels were not. The majority of the internals components containing crevices, such as baffle and former assemblies and fasteners, are fabricated from austenitic stainless steel alloys and hence not anticipated to be susceptible to pitting.

Only limited information is available on the effect of hydrogen on crevice corrosion. However, there are sufficient data to indicate that hydrogen overpressure plays an important role in the control of crevice corrosion in PWRs by minimizing the adverse effects of oxygen through recombination [4-35]. A literature survey of stainless steel corrosion revealed that the presence of crevice conditions would not greatly increase the corrosion rate in a PWR environment due to the low oxygen levels present during reactor operation [4-34, 4-37 through 4-41]. Furthermore, in addition to the control of radiolytic dissolved oxygen in the regions of the core by excess dissolved hydrogen, stagnant (crevice) regions are protected by [4-14]: (1) diffusion of hydrogen from the bulk coolant, (2) enhanced recombination rates from the effects of the crevice on the radiation quality, and (3) rapid reaction of the oxygen with the materials of construction leaving an excess of hydrogen in the stagnant areas. Therefore, it is concluded that maintaining a concentration of hydrogen on the order of 25 to 50 cc/kg of water is expected to provide adequate protection against crevice corrosion in PWRs both through the current operating license and into life extension.

No further evaluation is required with respect to corrosion for the PWR reactor internals components fabricated form austenitic stainless steel or nickel-base alloys. A license renewal applicant intending to take credit for these conclusions is responsible for the review/evaluation of plant-specific features, including appropriate CLB documents/information in order to assure that the above assumptions and criteria are met.

4.10 FATIGUE

4.10.1 Mechanism Description

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

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

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

4.10.2 Determination Of Significance

PWR RPV internals components are subject to fluctuating loads with a variety of frequencies, ranging from the relatively infrequent (refueling) to the relatively frequent (flow-induced vibration). Components which undergo significant thermal and seismic events are potentially susceptible to low-cycle fatigue damage. Those components which are subject to significant dynamic loads associated with flow-induced vibration are potentially susceptible to high-cycle fatigue damage. The design bases for many PWR internals components have included fatigue evaluations, both for those designed to the ASME Code, Section III, Subsection NG and those earlier plants with a more ad hoc fatigue design basis (Table 3.1). However, only the Subsection NG plants have available systematic and detailed fatigue analyses of component low-cycle and high-cycle fatigue usage.

ASME Code, Section III fatigue design procedures utilize a design fatigue curve, which is a plot of alternating stress range (S_a) versus the number of cycles to failure (N). The design fatigue curve is based on the un-notched fatigue properties of the material, modified by reduction factors that account for various geometric and moderate environmental effects. The fatigue usage factor (u) is defined by Miner's Rule as the summation of the damage over the total number of design basis transient types (I), as given by the ratio of expected cycles of that type (n_i) to the allowable number of cycles (N_i) for the stress ranges associated with that transient:

$$u = \sum_{i=1}^{I} \frac{n_i}{N_i}$$

For ASME Code design acceptance, the usage factor calculated in this manner cannot exceed unity (1.0) for the design lifetime of the component.

The conservatism in ASME Code fatigue calculations stems from two sources. First, the design fatigue curves contain either a factor of two on stress range or a factor of twenty on the number of cycles to failure, depending upon which is controlling. Second, a substantial margin is also expected to exist because of conservatisms in the magnitude and frequency of occurrence assumed for the various design basis transients [4-1, 4-3, 4-42, and 4-43]. An additional source of conservatism with respect to high-cycle fatigue for internals components in operating plants can be derived on the basis of the fatigue curves for typical internals materials. It is characteristic of these curves that, as the number of cycles-to-failure increases beyond 10^6 cycles, the magnitude of the alternating stress range becomes less sensitive to the number of loading cycles. The stress range is approaching the endurance limit of the material. Typical PWR internals vibration frequencies are in the range of 5 to 10 Hz, so that an operating plant will accumulate 10^9 - 10^{10} fatigue cycles in less than 40 years. In practical terms this means that, in the absence of changes in loading or configuration, internals components which have not experienced high-cycle fatigue damage during the original license period are unlikely to experience high-cycle fatigue damage during the license renewal term.

This insensitivity to stress range for high-cycle loading can also be the basis for initial design, and helps to explain some of the differences in the approaches that Westinghouse, Combustion Engineering, and Babcock & Wilcox have used to evaluate low cycle and high cycle fatigue of PWR internals components. Westinghouse and Combustion Engineering have used conservative calculations of bounding design transients and subsequent design basis stresses to estimate low cycle fatigue accumulation for the specified transients. The evaluation of high cycle fatigue has been based largely on results of hot functional tests. The rationale for the latter is that a component with high cycle fatigue susceptibility (i.e., the stresses caused by the high-cycle loading are well above the endurance limit of the material) will be identified during hot functional testing which induce higher flow loads without the resistance of the fuel assemblies. Any high cycle fatigue issues which have been identified by hot functional testing have either required subsequent design or operational modifications, or analyses to demonstrate the acceptability of the observed behavior. Because of this reliance on testing, high cycle fatigue usage estimates of Westinghouse and Combustion Engineering internals components are realistically small, and overall fatigue usage estimates tend to be dominated by conservative calculations of low cycle fatigue usage.

For Babcock & Wilcox plants, both low cycle fatigue and high cycle fatigue usage factors have been calculated [4-44 and 4-45], using standard ASME Code procedures. High cycle fatigue usage factors were calculated using extensions of the design curves in the 1974 edition of Section III, Subsection NG of the ASME Code. The extensions made by Babcock & Wilcox to 10^{12} cycles were subsequently found to be significantly lower (and, hence, more

conservative) than the fatigue curves in the 1983 edition of the ASME Code, whose ranges extend to 10^{11} cycles [4-3].

As a result, the combined low-cycle and high-cycle fatigue usage estimates for Westinghouse, Combustion Engineering, and Babcock & Wilcox plants are all expected to be conservatively high. For Westinghouse and Combustion Engineering plants, the predominant conservatisms are in the low-cycle fatigue calculations, due to the design basis transient definitions. For Babcock & Wilcox plants, the major conservatisms are in the high-cycle fatigue calculations, due to the definition of cyclic failure limits. It should be pointed out that these conservatisms are in addition to the ASME Code factor of two on stress range or twenty on cycles to failure inherent in the Code fatigue curves.

For license renewal evaluation purposes, fatigue will be considered to be a nonsignificant age-related degradation mechanism for a component when two criteria are met: (1) The component must have no history of fatigue damage and (2) the fatigue usage factor must be below any threshold of concern. Any component that does not meet these two criteria will be considered a fatigue-critical component which requires further evaluation for license renewal purposes. For the definition of fatigue-critical, the fatigue usage factor threshold of concern is 0.4.

For those plants with PWR RPV internals components with an explicit Section III fatigue design basis, or its equivalent, fatigue-critical internals components can be identified, for license renewal purposes, from plant design stress reports or hot functional test data reports. For earlier plants whose internals components did not have an explicit fatigue design basis, the principles of similitude apply. Similitude demonstration is based upon a number of factors, foremost of which are similarity of geometry and similarity of component operating history. For example, Babcock & Wilcox [4-3] used the explicit fatigue analysis results for their 205 FA plant RPV internals and their similarity to geometries and operating conditions for 177 FA plant RPV internals to assess fatigue damage in an early plant. The fatigue-critical components were identified as components with an original fatigue usage factor greater than 0.67. This list consisted of the upper core support cylinder, lower core support cylinder, baffle-former bolts, plenum cylinder/upper grid bolts, upper grid and column weldment/upper grid bolts.

These components were then reanalyzed to include a realistic (RMS) evaluation of cyclic amplitudes, with the new fatigue usage factors found to be considerably lower than the originally-calculated fatigue usage factors, as shown below:

B&W Internals Components	Usage Factors	
	Original	New Calculation
Core Support Shield	.86	.13
Core Barrel	.98	.21
Former Bolts	.68	.17
Plenum Cylinder / Upper Grid Bolts	.73	.21
Upper Grid Rib Section	.72	.34
Upper Grid Assembly / Upper Grid Bolts	.73	.03

Note that, as a result of the recalculations, the maximum fatigue usage factors are well below any threshold of concern (e.g., 0.4) that might lead to concern about an extended license period. Furthermore, evaluation of operating histories and component geometries at early plants revealed that these same conclusions apply to B&W internals components with no explicit Section III fatigue design basis.

Fatigue-critical internals components have also been identified for Westinghouse and Combustion Engineering plants, using design stress report and hot functional test data for later plants. For Westinghouse units, the fatigue-critical components were determined to be [4-1]:

- Upper Support Plate
- Upper Support Column Bolts
- Lower Core Plate (in particular, the minimum-thickness ligament of the perforated region)
- Lower Support Columns
- Core Barrel Nozzles (weldments)
- Baffle-Former Assembly Bolts
- Upper Core Plate Alignment Pins (weldments)
- RCCA Guide Tubes (weldments)

A screening fatigue usage factor threshold of 0.25 for a design life of 40 years was used as the basis for the selection of Westinghouse fatigue-critical components.

For Combustion Engineering units, a combination of design stress report review, hot functional test data and engineering expert judgement led to the following list of fatigue-critical components:

- CEA Shroud Bolts
- Core Shroud Assembly Bolts
- Core Shroud Tie Rods
- Core Support Plate
- Core Support Column Bolts

Therefore, with the exception of those PWR RPV internals components identified in Table 4.7 as fatigue-critical -- based upon a review of calculated fatigue usage factors for internals components designed to ASME Section III, Subsection NG; a review of hot functional test data; and a comparison of geometric and operating similarities -- fatigue is not a significant age-related degradation mechanism for PWR internals components. Fatigue-critical PWR internals components require further evaluation as provided in Section 5. A license renewal applicant intending to take credit for these conclusions is responsible for the review/evaluation of plant-specific features, including appropriate CLB documents/information in order to assure that the above assumptions and criteria are met.

Table 4.7

FATIGUE SUMMARY

for each STRUCTURAL ASSEMBLY GROUPING

Key: ☐ Degradation not significant
■ Degradation potentially significant

UPPER INTERNALS ASSEMBLY

Westinghouse		Combustion Engineering		Babcock & Wilcox (2)	
Upper Internals Assembly		**Upper Guide Structure Assembly**		**Plenum Assembly**	
1. Upper Support Plate	■	1. Upper Guide Structure Support Plate	☐	1. Plenum Cover and Plenum Cylinder	☐
2. RCCA Guide Tube Assemblies	☐	2. CEA Shroud Assemblies	☐	2. CRA Guide Tube Assemblies	☐
RCCA Guide Tubes (1)	■	CEA Shrouds	☐	CRA Guide Tubes	☐
RCCA Guide Tube Bolts	☐	CEA Shroud Bolts	■	CRA Guide Tube Bolts	☐
RCCA Guide Tube Support Pins	☐				
3. Upper Support Columns	☐	3. ----------		3. ----------	
Upper Support Column Bolts	■				
4. Upper Core Plate	☐	4. Fuel Alignment Plate	☐	4. Upper Grid Assembly	☐
Fuel Pins	☐			Upper Grid Rib Section	☐
				Upper Grid Assembly Bolts	☐
				Fuel Guide Pads	☐

CORE SUPPORT ASSEMBLY

Core Barrel and Baffle/Former Assembly		**Core Support Barrel and Core Shroud Assembly**		**Core Support Assembly**	
1. Core Barrel	☐	1. Core Support Barrel	☐	1. Core Support Shield	☐
Upper Core Barrel Flange	☐	Core Support Barrel Upper Flange	☐	Core Support Shield Flange	☐
Core Barrel Nozzles	■				
2. ----------		2. ----------		2. Vent Valve Assemblies	☐
3. Baffle-Former Assembly	☐	3. Core Shroud Assembly	■	3. Core Barrel Assembly	☐
Baffle / Former Assembly Bolts	■	Core Shroud Assembly Bolts	☐	Core Barrel Bolts	☐
Baffle / Former Assembly Baffles	☐	Core Shroud Tie Rods	■	Baffle / Former Plates	☐
				Baffle / Former Bolts	☐
4. Upper Core Plate Alignment Pins	■	4. Fuel Alignment Plate Guide Lugs	☐	4. ----------	

LOWER INTERNALS ASSEMBLY

Lower Internals Assembly		**Core Support Assembly**		**Lower Grid Assembly**	
1. Lower Core Plate	■	1. Core Support Plate	■	1. Lower Grid Top Rib Section	☐
2. Fuel Pins	☐	2. Fuel Alignment Pins	☐	2. Fuel Guide Pads	☐
3. Lower Support Plate	☐	3. Lower Support Structure Beam Assemblies	☐	3. Lower Grid Bottom Rib Weldment	☐
4. Lower Support Columns	■	4. Core Support Columns	■	4. Lower Grid Assembly Support Posts	☐
Lower Support Column Bolts	☐	Core Support Column Bolts	☐	Lower Grid Assembly Bolts	☐
5. Radial Keys and Clevis Inserts	☐	5. Core Support Barrel Snubber Assemblies	☐	5. Lower Grid Cylinder Guide Blocks	☐

NOTES: (1) Fatigue concern is at the RCCA Guide Tube Welds. (2) No B&W fatigue concerns based on re-evaluation.

4.11 DEGRADATION MECHANISM ASSESSMENT SUMMARY

This section contains a summary of the age-related degradation mechanisms investigated in this report. Those mechanisms that could cause potentially significant degradation to the PWR reactor internals components which are covered by this report are listed below. The component parts carried forward to Sections 5 and 6 are listed in Table 4.8. A license renewal applicant intending to take credit for these conclusions is responsible for the review/evaluation of plant-specific features for each of the components covered in this report, including appropriate CLB documents/information, in order to assure that the assumptions and criteria are met.

4.11.1 Irradiation Embrittlement

Irradiation embrittlement is a potentially significant age-related degradation mechanism for those PWR reactor internals components fabricated from austenitic stainless steel and nickel-base alloy materials which experience neutron fluences in excess of 1×10^{20} n/cm² (E > 1 MeV). Further evaluation for those components projected to exceed this screening threshold is provided in Section 5. For PWR reactor internals components not exceeding this threshold value, irradiation embrittlement is nonsignificant (Section 4.1).

4.11.2 Stress Corrosion Cracking

Initiation and propagation of stress corrosion cracking requires three factors to be present: (1) a susceptible material, (2) a corrosive environment and (3) the presence of tensile stresses. The only PWR reactor internals components for which stress corrosion cracking is potentially significant are Westinghouse RCCA guide tube support pins manufactured from Alloy X-750, Combustion Engineering CEA shroud bolts and B&W core barrel bolts, both manufactured from Alloy A-286. Further evaluation for these components is provided in Section 5. The degradation sustained from stress corrosion cracking is nonsignificant for all other PWR reactor internals components covered by this report (Section 4.2).

4.11.3 Irradiation-assisted Stress Corrosion Cracking

IASCC is a nonsignificant age-related degradation mechanism for PWR RPV internals components provided that the PWR primary coolant is maintained such that oxygen concentrations are < 5 ppb and halogen concentrations are < 150 ppb, the fluence level is less than 5×10^{20} n/cm² (low stress level) or 1×10^{20} n/cm² (moderate to high stress stress level) and operating stresses are within the design envelope. Further evaluation of components that do not meet the above criteria is required and is provided in Section 5 (Section 4.3).

4.11.4 Erosion and Erosion/Corrosion

The degradation sustained from erosion and erosion/corrosion is nonsignificant to all PWR reactor internals components covered by this report (Section 4.4).

4.11.5 Creep

The degradation sustained from creep is nonsignificant to all PWR reactor internals components covered by this report (Section 4.5).

4.11.6 Stress Relaxation

Stress relaxation is a potentially significant age-related degradation mechanism to components with substantial preloads, such as torqued bolts, because the maintenance of an adequate preload is important to their functionality. Further evaluation of these components is provided in Section 5. For all other PWR reactor internals components which do not depend on preload for functionality, stress relaxation is nonsignificant (Section 4.6).

4.11.7 Wear

Wear is a potentially significant age-related degradation mechanism at the interfaces of components whose relative motion is not completely restrained. Further evaluation of these components is provided in Section 5. For all other PWR reactor internals components, wear is nonsignificant (Section 4.7).

4.11.8 Thermal Aging

The components to which thermal aging could be potentially significant are those containing cast austenitic stainless steel (CASS) and martensitic stainless steel subcomponents in the B&W vent valve assemblies. Thermal aging is potentially significant for those CASS PWR reactor internals components exceeding the threshold criteria provided in Section 4.8 and further evaluation is provided in Section 5. Thermal aging embrittlement is also a potentially significant age-related degradation mechanism for the martensitic stainless steel subcomponents in the B&W vent valve assemblies. Further evaluation of these vent valve assemblies is provided in Section 5. For all other PWR reactor internals components covered by this report, thermal aging is nonsignificant (Section 4.8).

4.11.9 Corrosion

The degradation sustained from corrosion is nonsignificant to all PWR reactor internals components covered by this report (Section 4.9).

4.11.10 Fatigue

Fatigue is a potentially significant age-related degradation mechanism only for PWR reactor internals components determined to be fatigue-critical. This determination has its basis in analysis, experiment and experience. A review of calculated fatigue usage factors for internals components designed to ASME Section III, Subsection NG, a review of hot functional test data, and a comparison of geometric and operating similarities served to identify the fatigue-critical internals components. Further evaluation of the fatigue-critical components is

provided in Section 5. For all other PWR reactor internals components covered by this report, fatigue is nonsignificant (Section 4.10).

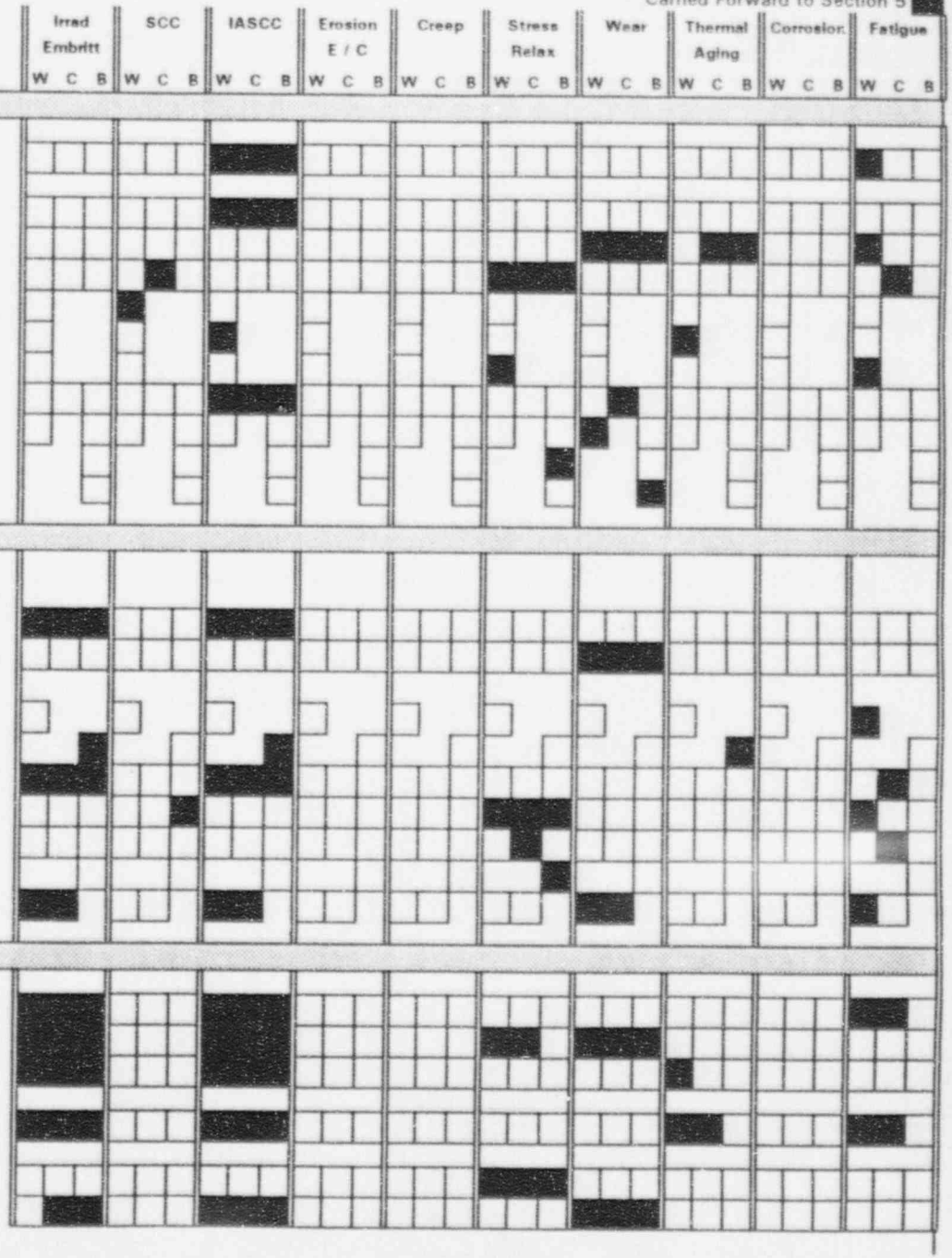
Table 4.8

DEGRADATION MECHANISM SUMMARY
IDENTIFYING COMPONENTS CARRIED FORWARD TO SECTION 5

for each STRUCTURAL ASSEMBLY GROUPING

Key: No Significant Degradation; Carried Forward to Section 5

WESTINGHOUSE	COMBUSTION ENGINEERING	BABCOCK & WILCOX
UPPER INTERNALS ASSEMBLY		
Upper Internals Assembly	**Upper Guide Structure Assembly**	**Plenum Assembly**
1. Upper Support Plate	1. Upper Guide Structure	1. Plenum Cover and Plenum Cylinder
2. RCCA Guide Tube Assemblies	2. CEA Shroud Assemblies	2. CRA Guide Tube Assemblies
RCCA Guide Tubes	CEA Shrouds	CRA Guide Tubes
RCCA Guide Tube Bolts	CEA Shroud Bolts	CRA Guide Tube Bolts
RCCA Guide Tube Support Pins		
3. Upper Support Columns	3. ----------	3. ----------
Upper Support Column Bolts		
4. Upper Core Plate	4. Fuel Alignment Plate	4. Upper Grid Assembly
Fuel Pins		Upper Grid Rib Section
		Upper Grid Assembly Bolts
		Fuel Guide Pads
CORE SUPPORT ASSEMBLY		
Core Barrel and Baffle/Former Assembly	**Core Support Barrel and Core Shroud Assembly**	**Core Support Assembly**
1. Upper Core Barrel	1. Core Support Barrel	1. Core Support Shield
Core Barrel Flange	Core Support Barrel Upper Flange	Core Support Shield Flange
Core Barrel Nozzles		
2. ----------	2. ----------	2. Vent Valve Assemblies
3. Baffle-Former Assembly	3. Core Shroud Assembly	3. Core Barrel Assembly
Baffle / Former Assembly Bolts	Core Shroud Assembly Bolts	Core Barrel Bolts
Baffle / Former Assembly Baffles	Core Shroud Tie Rods	Baffle / Former Plates
		Baffle / Former Bolts
4. Upper Core Plate Alignment Pins	4. Fuel Alignment Plate Guide Lugs	4. ----------
LOWER INTERNALS ASSEMBLY		
Lower Internals Assembly	**Core Support Assembly**	**Lower Grid Assembly**
1. Lower Core Plate	1. Core Support Plate	1. Lower Grid Top Rib Section
2. Fuel Pins	2. Fuel Alignment Pins	2. Fuel Guide Pads
3. Lower Support Plate	3. Lower Support Structure Beam Assemblies	3. Lower Grid Bottom Rib Weldment
4. Lower Support Columns	4. Core Support Columns	4. Lower Grid Assembly Support Posts
Lower Support Column Bolts	Core Support Column Bolts	Lower Grid Assembly Bolts
5. Radial Keys and Clevis Inserts	5. Core Support Barrel Snubber Assemblies	5. Lower Grid Cylinder Guide Blocks



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5.0 EFFECTIVE PROGRAMS FOR POTENTIALLY SIGNIFICANT AGE-RELATED DEGRADATION

In the previous section of this report, the age-related degradation mechanisms that could effect the specific PWR RPV internals components were described and their potential significance to the continued safety function performance of these components throughout the license renewal term was evaluated. Specific assumptions and criteria were provided to enable the license renewal applicant to determine whether a particular age-related degradation mechanism/component combination requires further evaluation (potentially significant), or whether a review of appropriate plant design, construction and operation history documents and records would confirm that the degradation mechanism does not affect the component's capability to perform its intended safety function throughout the license renewal term.

This assessment reviewed age-related degradation mechanisms that have been derived from the available literature, in particular the Westinghouse [5-1], Combustion Engineering [5-2], and Babcock and Wilcox [5-3] plant-specific reports on PWR RPV internals as well as a review/evaluation of component service experience, relevant laboratory data, and related experience from other industries [5-4]. The set consists of the following:

1. Irradiation Embrittlement.
2. Stress Corrosion Cracking (SCC).
3. Irradiation-assisted Stress Corrosion Cracking (IASCC).
4. Erosion and Erosion/Corrosion.
5. Creep.
6. Stress Relaxation.
7. Wear.
8. Thermal Embrittlement
9. Corrosion.
10. Fatigue.

The significance of each of these age-related degradation mechanisms was evaluated for each of the PWR RPV internals components being evaluated in this report.

In this section those combinations of age-related degradation mechanisms and components that require further evaluation are re-examined in terms of the capability of effective programs for maintenance, inservice inspection, surveillance, testing and analytical assessment to manage the effects of the potentially significant degradation. Combinations of mechanisms and components for which generic program elements effectively manage the age-related degradation are considered to be adequately addressed.

A license renewal applicant intending to take credit for an effective program is responsible for the review/evaluation of appropriate plant-specific features, including appropriate CLB documents/information, in order to assure that the program elements required to manage the

effects of potentially significant age-related degradation, or their justified equivalent, continue being used at their plant. Plant-specific evaluations are required for any age-related degradation/component issues for which the license renewal applicant is unable to demonstrate that the generic stipulated program elements are committed for use, or the basis for the conclusions are applicable, at their plant. Recommendations for plant-specific aging management options for these issues are provided in Section 6 of this report.

An age-related degradation mechanism is defined to be significant for a component if, when allowed to continue without an effective program, the capability of the component to perform its intended safety function throughout the license renewal term would be compromised. The potential significance of an age-related degradation mechanism was determined in Section 4 by examining the component design features (Section 3.1), the component design bases (Section 3.2), its operation and maintenance histories (Section 3.3), and its susceptibility to the degradation mechanism being considered. If it could be shown that the component is either not susceptible, or is susceptible to such a small degree that the component's safety function is maintained throughout the license renewal term, then the component/degradation mechanism combination is not significant.

The potentially significant age-related degradation mechanism/component or subcomponent combinations identified in Section 4 and addressed in the following subsections are summarized in Table 5.1.

Table 5.1
Components / Subcomponents Requiring Section 5 Evaluation

Degradation Mechanism	W	CE	B&W
Irradiation Embrittlement	- Upper Core Barrel - Baffle / Former Assembly - Upper Core Plate Alignment Pins - Lower Core Plate - Fuel Pins - Lower Support Plate - Lower Support Columns	- Core Support Barrel - Core Shroud Assembly - Fuel Alignment Plate Guide Lugs - Core Support Plate - Fuel Alignment Pins - Lower Support Structure Beam Assemblies - Core Support Columns - Core Support Barrel Snubber Assemblies	- Core Support Shield - Vent Valve Assemblies - Core Barrel Assembly - Lower Grid Top Rib Section - Fuel Guide Pads - Lower Grid Bottom Rib Weldment - Lower Grid Assembly Support Posts - Lower Grid Cylinder Guide Blocks
Stress Corrosion Cracking (SCC)	- RCCA Guide Tube Support Pins	- CEA Shroud Bolts	- Core Barrel Bolts
Irradiation-assisted Stress Corrosion Cracking (IASCC)	- ALL	- ALL	- ALL
Stress Relaxation	- RCCA Guide Tube Bolts - Upper Support Column Bolts - Baffle / Former Assembly Bolts - Fuel Pins - Lower Support Column Bolts	- CEA Shroud Bolts - Core Shroud Assembly Bolts - Core Shroud Tie Rods - Fuel Alignment Pins - Core Support Column Bolts	- CRA Guide Tube Bolts - Upper Grid Assembly Bolts - Core Barrel Bolts - Baffle / Former Bolts - Lower Grid Assembly Bolts
Wear	- RCCA Guide Tubes - Core Barrel Flange - Upper Core Plate Alignment Pins - Fuel Pins - Radial Keys and Clevis Inserts	- CEA Shrouds - Fuel Alignment Plate - Core Support Barrel Upper Flange - Fuel Alignment Plate Guide Lugs - Fuel Alignment Pins - Core Support Barrel Snubber Assemblies	- CRA Guide Tubes - Fuel Guide Pads - Core Support Shield Flange - Lower Grid Cylinder Guide Blocks
Thermal Embrittlement (CASS)	- Upper Support Columns - Lower Support Plate - Lower Support Columns	- CEA Shrouds - Core Support Columns	- CRA Guide Tubes - Vent Valve Assemblies
Fatigue	- Upper Support Plate - RCCA Guide Tubes - Upper Support Column Bolts - Core Barrel Nozzles - Baffle / Former Assembly Bolts - Upper Core Plate Alignment Pins - Lower Core Plate - Lower Support Columns	- CEA Shroud Bolts - Core Shroud Assembly - Core Shroud Tie Rods - Core Support Plate - Core Support Columns	- NONE (See Section 4.10)

5.1 IRRADIATION EMBRITTLEMENT

Neutron irradiation embrittlement was identified in Section 4.1 as a potentially significant age-related degradation mechanism for Core Support Assembly and Lower Internals Assembly components of PWRs. A list of the affected internals components is given below.

W	CE	B&W
- Upper Core Barrel	- Core Support Barrel	- Core Support Shield
- Baffle / Former Assembly	- Core Shroud Assembly	- Vent Valve Assemblies
- Upper Core Plate Alignment Pins	- Fuel Alignment Plate Guide Lugs	- Core Barrel Assembly
- Lower Core Plate	- Core Support Plate	- Lower Grid Top Rib Section
- Fuel Pins	- Fuel Alignment Pins	- Fuel Guide Pads
- Lower Support Plate	- Lower Support Structure Beam Assemblies	- Lower Grid Bottom Rib Weldment
- Lower Support Columns	- Core Support Columns	- Lower Grid Assembly Support Posts
	- Core Support Barrel Snubber Assemblies	- Lower Grid Cylinder Guide Blocks

Examination Category B-N-3 of the ASME Code Section XI, Subsection IWB (See IWB-2500-1) identifies inservice examination requirements for accessible surfaces of PWR reactor vessel core support structures that can be removed from the reactor vessel for examination. The required examination method is visual (VT-3) and the examination interval depends on the choice of inspection program, either Program A or Program B (See Section 3.2), as described in Article IWB-2000. Acceptance standards for the examinations are given in IWB-3520.2.

For irradiation embrittlement, the relevant conditions related to cracked or fractured parts are of most interest, since the anticipated manifestation of neutron irradiation embrittlement damage is a lower resistance to crack initiation. As a result of VT-3 standards with respect to demonstrated character recognition height and minimum distance from the examiner's eye to the object surface, adequate detectability of surface flaws of significant length is assured.

Therefore, visual (VT-3) inservice examination of core support assembly and lower internals assembly PWR RPV internals components, in accordance with the requirements for Examination Category B-N-3 of Subsection IWB, Section XI, is able to manage the effects of irradiation embrittlement, provided that such components are accessible, or can be rendered accessible by the removal of the core and/or other internals, for the examination.

A license renewal applicant intending to take credit for this effective program is responsible for the review/evaluation of plant-specific documentation, in order to assure that the program elements required to manage the effects of the potentially significant irradiation embrittlement, or the justified equivalent, are committed for use at their plant. If the elements of an adequate

visual (VT-3) inservice examination program are not in place, or if the affected component is not accessible, or cannot be rendered accessible by the removal of the core and/or other internals, for the examination, then alternatives for managing the effects of irradiation embrittlement are provided in Section 6.

5.2 STRESS CORROSION CRACKING

Stress corrosion cracking (SCC) was identified in Section 4.2 as a potentially significant age-related degradation mechanism for components which are fabricated from Alloy A-286 or Alloy X-750 material. The affected components are listed below.

W	CE	B&W
- RCCA Guide Tube Support Pins	- CEA Shroud Bolts	- Core Barrel Bolts

Examination Category B-N-3 of Section XI, Subsection IWB, provides requirements for the visual (VT-3) examination of "removable core support structures" of PWRs. These requirements refer to the relevant conditions defined in IWB-3520.2, which include "loose, missing, cracked, or fractured parts, bolting, or fasteners." If any relevant condition is detected, IWB-3142 provides options for correcting the relevant condition, such as: (*i*) acceptance by supplemental surface and/or volumetric examination, in order to characterize the indication more accurately; (*ii*) acceptance by analytical evaluation, which may involve more frequent examination of the item; or (*iii*) acceptance by replacement of the item.

For the case of Alloy A-286 core barrel bolts at Babcock & Wilcox plants, a limited number of Alloy A-286 core barrel bolt failures have been observed that may be attributable to SCC. In all cases where such failures have occurred, the affected and potentially susceptible bolts have been replaced by bolts that have been designed to maintain stress levels below the SCC threshold for initiation or that have been fabricated from materials less susceptible to SCC, or both. In addition, Babcock & Wilcox has recommended periodic inspection programs to plant owners [5-5] in order to monitor potential age-related degradation associated with SCC of core barrel bolts.

CEA shroud bolts for Combustion Engineering plant are also fabricated of Alloy A-286 material. In this case, since the upper internals are removed at each refueling outage, visual inspection in accordance with the procedures of the ASME Code, Section XI, Subsection IWB provides assurance that the effects of potential SCC of the CEA shroud bolts are detected and the age-related degradation is effectively managed [5-6].

RCCA guide tube support pins in Westinghouse plants are fabricated from Alloy X-750. Some SCC and eventual fracture of these pins have been observed in service and attributed to the combination of high stress and a susceptible material condition. In the majority of the affected plants, these Alloy X-750 support pins have been replaced by pins designed to reduce

the applied stresses below the SCC threshold and fabricated from a material with a heat treatment to reduce susceptibility. All of the affected utilities have developed and put procedures in place to detect and control any incipient SCC degradation of these components [5-7 through 5-9].

Therefore, visual (VT-3) inservice examination of Alloy A-286 core barrel bolts in B&W plants, Alloy A-286 CEA shroud bolts in Combustion Engineering plants, and Alloy X-750 RCCA guide tube support pins, in accordance with the requirements for examination category B-N-3 of Subsection IWB, Section XI, is able to manage the effects of stress corrosion cracking, provided that such components are accessible, or can be rendered accessible by the removal of the core and/or other internals, for the examination.

A license renewal applicant intending to take credit for this effective program is responsible for the review/evaluation of plant-specific documentation, in order to assure that the program elements required to manage the effects of the potentially significant stress corrosion cracking, or the justified equivalent, are committed for use at their plant. If the elements of an adequate visual (VT-3) inservice examination program are not in place, or if the affected component is not accessible, or cannot be rendered accessible by the removal of the core and/or other internals, for the examination, then alternatives for managing the effects of stress corrosion cracking are provided in Section 6.

5.3 IRRADIATION-ASSISTED STRESS CORROSION CRACKING

Irradiation-assisted stress corrosion cracking (IASCC) was identified in Section 4.3 as a potentially significant, synergistic age-related degradation mechanism for all PWR RPV internals components.

W	CE	B&W
ALL	ALL	ALL

Examination Category B-N-3 of Section XI, Subsection IWB, provides requirements for the visual (VT-3) examination of "removable core support structures" of PWRs. These requirements refer to the relevant conditions defined in IWB-3520.2, which include "loose, missing, cracked, or fractured parts, bolting, or fasteners." If any relevant condition is detected, IWB-3142 provides options for correcting the relevant condition, such as: (*i*) acceptance by supplemental surface and/or volumetric examination, in order to characterize the indication more accurately; (*ii*) acceptance by analytical evaluation, which may involve more frequent examination of the item; or (*iii*) acceptance by replacement of the item.

RCCA guide tube support pins in Westinghouse plants are fabricated from Alloy X-750. Some SCC and eventual fracture of these pins have been observed in service and attributed to the combination of high stress and a susceptible material condition. The effect of radiation may have also led to enhanced crack-tip strain rates that could add the synergy of radiation to the combination of high stress and material conditionn causing these incidences. In the

of the affected plants, those Alloy X-750 support pins associated with the high stress SCC threshold have been replaced by pins designed to reduce the applied stresses below the SCC threshold and fabricated from a material with a heat treatment to reduce susceptibility. Therefore, the SCC threshold for such components has been raised from 1×10^{20} to 5×10^{20} n/cm^2. All of the affected utilities have developed and put procedures in place to detect and control any incipient SCC degradation, which includes IASCC, of these components [5-7 through 5-9].

The replacement Alloy X-750 core barrel bolts in the Babcock & Wilcox plants have been designed to maintain stress levels below the SCC threshold for initiation and are discussed here only to insure complete consideration of all Alloy X-750 PWR RPV internals components within the scope of this report. As discussed in Section 3.3, Babcock & Wilcox has recommended periodic inspection programs to plant owners [5-5] in order to monitor potential age-related degradation associated with SCC of core barrel bolts, which also includes monitoring of these replacement bolts. No indication of IASCC (or SCC) has been observed in these replacement bolts.

Visual (VT-3) inservice examination of Alloy X-750 Westinghouse RCCA guide tube support pins and B&W replacement core barrel bolts, in accordance with the requirements for examination category B-N-3 of Subsection IWB, Section XI, has been shown to manage the effects of irradiation-assisted stress corrosion cracking. Therefore, VT-3 inservice examination of other PWR RPV internals components will also be sufficient to manage the effects of IASCC. A license renewal applicant intending to take credit for this effective program is responsible for the review/evaluation of plant-specific documentation, in order to assure that the program elements required to manage the effects of the potentially significant irradiation-assisted stress corrosion cracking, or the justified equivalent, are committed for use at their plant.

5.4 STRESS RELAXATION

Stress relaxation was identified as a potentially significant age-related degradation mechanism for PWR RPV internals components that require adequate preload in order to perform their intended safety function in Section 4.6. The components in this category are listed below.

W	CE	B&W
- RCCA Guide Tube Bolts	- CEA Shroud Bolts	- CRA Guide Tube Bolts
- Upper Support Column Bolts	- Core Shroud Assembly Bolts	- Upper Grid Assembly Bolts
- Baffle / Former Assembly Bolts	- Core Shroud Tie Rods	- Core Barrel Bolts
- Fuel Pins	- Fuel Alignment Pins	- Baffle / Former Bolts
- Lower Support Column Bolts	- Core Support Column Bolts	- Lower Grid Assembly Bolts

Examination Category B-N-3 of Section XI, Subsection IWB, provides requirements for the visual (VT-3) examination of accessible surfaces of PWR core support structures that can be removed from the reactor vessel. These requirements refer to the relevant conditions defined in IWB-3520.2, which include "loose, missing, cracked, or fractured parts, bolting, or fasteners." Since the manifestation of excessive stress relaxation is expected to be loose, cracked or missing bolts or fasteners, the VT-3 examination is adequate for the detection of significant stress relaxation and loss of preload. If any relevant condition is identified, IWB-3142 provides options for the timely correction of the problem, such as: (*i*) acceptance by supplemental surface and/or volumetric examination, in order to further characterize the condition; (*ii*) acceptance by corrective measures (e.g., reestablishing the preload) or repairs; and (*iii*) acceptance by replacement of the item.

For those few cases where the relevant conditions of Examination Category B-N-3 may be unable to detect significant stress relaxation and loss of preload because the assurance of functional capability cannot be guaranteed by detection of loose, missing, cracked, or fractured parts, bolting, or fasteners, alternatives for managing the effects of age-related degradation are described in Section 6. Baffle-former assembly bolts in Westinghouse and Babcock & Wilcox plants, and core shroud assembly bolts in Combustion Engineering plants, have been deemed to be in this category.

With these exceptions, visual (VT-3) inservice examination of PWR RPV internals components (in accordance with the requirements of Examination Category B-N-3 of Subsection IWB, Section XI) that require adequate preload in order to perform their intended safety function, is able to manage the effects of stress relaxation, provided that such components are accessible, or can be rendered accessible by the removal of the core and/or other internals, for the examination.

A license renewal applicant intending to take credit for this effective program is responsible for the review/evaluation of plant-specific documentation, in order to assure that the program elements required to manage the effects of the potentially significant stress relaxation, or the justified equivalent, are committed for use at their plant. If the elements of an adequate visual (VT-3) inservice examination program are not in place, or if the effected component is not accessible, or cannot be rendered accessible by the removal of the core and/or other internals, for the examination, then alternatives for managing the effects of stress relaxation are provided in Section 6.

5.5 WEAR

Wear was identified as a potentially significant age-related degradation mechanism for PWR RPV internals components which may have relative motion with adjacent surfaces in Section 4.7. The affected components are listed below.

W	CE	B&W
- RCCA Guide Tubes	- CEA Shrouds	- CRA Guide Tubes
- Core Barrel Flange	- Fuel Alignment Plate	- Fuel Guide Pads
- Upper Core Plate Alignment Pins	- Core Support Barrel Upper Flange	- Core Support Shield Flange
- Fuel Pins	- Fuel Alignment Plate Guide Lugs	- Lower Grid Cylinder Guide Blocks
- Radial Keys and Clevis Inserts	- Fuel Alignment Pins	
	- Core Support Barrel Snubber Assemblies	

Examination Category B-N-3 of Section XI, Subsection IWB, provides requirements for the visual (VT-3) examination of accessible surfaces of PWR core support structures that can be removed from the reactor vessel. These requirements refer to the relevant conditions defined in IWB-3520.2, which include "wear of mating surfaces that may lead to loss of function." Since the manifestation of excessive wear is expected to be wear of mating surfaces that may lead to loss of function, the VT-3 examination is adequate for the detection of significant wear. If any relevant condition is identified, IWB-3142 provides options for the timely correction of the condition, such as acceptance by corrective measures or repairs.

Wear between two surfaces subject to relative motion is readily detectable by visual inspection long before the effects of wear begin to compromise the structural integrity or function of the component. If it is determined that excessive wear has taken place, by visual inspection or Code-defined alternate NDE means, subsequential wear evaluation can be performed to determine the need for refurbishment or repair of the worn areas.

Therefore, visual (VT-3) inservice examination of PWR reactor internals components that may have relative motion with respect to adjacent surfaces, in accordance with the requirements for examination category B-N-3 of Subsection IWB, Section XI, is able to manage the effects of wear. A license renewal applicant intending to take credit for this effective program is responsible for the review/evaluation of plant-specific documentation, in order to assure that the program elements required to manage the effects of the potentially significant wear, or the justified equivalent, are committed for use at their plant.

5.6 THERMAL AGING

Thermal aging was identified as a potentially significant age-related degradation mechanism for some PWR RPV internals components fabricated from cast austenitic stainless steel or their martensitic stainless steel subcomponents in Section 4.8. The affected components are listed below:

W	CE	B&W
- Upper Support Columns	- CEA Shrouds	- CRA Guide Tubes
- Lower Support Plate	- Core Support Columns	- Vent Valve Assemblies
- Lower Support Columns		

Examination Category B-N-3 of Section XI, Subsection IWB, provides requirements for the visual (VT-3) examination of accessible surfaces of PWR core support structures that can be removed from the reactor vessel. These requirements refer to the relevant conditions defined in IWB-3520.2, which include "loose, missing, cracked, or fractured parts." Since manifestation of excessive thermal aging is expected to be a reduced resistance to crack initiation, the VT-3 examination is adequate for the detection of significant thermal aging. If any relevant condition is identified, IWB-3142 provides options for the timely correction of the condition, such as: (*i*) acceptance by supplemental surface and/or volumetric examination, in order to characterize the indication more accurately; (*ii*) acceptance by analytical evaluation, which may involve more frequent examination of the item; and (*iii*) acceptance by replacement of the item.

The vent valve assemblies in Babcock & Wilcox plants undergo detailed inspection and testing each time the reactor vessel closure head is removed. The vent valves are visually inspected during normal in-service inspection. Therefore, current inspection and testing programs are adequate to manage any thermal aging of cast austenitic stainless steel vent valve assemblies, including the martensitic stainless steel subcomponents in Babcock & Wilcox plants.

CEA shrouds in Combustion Engineering plants and CRA guide tubes in Babcock & Wilcox plants provide vertical core support and must maintain sufficient integrity to ensure that control rod assemblies can be inserted into the core. No effects of thermal aging have been observed to date for these components; nevertheless, CEA shrouds (CE) and CRA guide tubes are accessible at each refueling and at ISI. Visual inspections performed at these intervals are adequate to detect any flaws which may have occurred in these components. Therefore, current inspection and flaw evaluation programs are adequate to manage any thermal aging of cast austenitic stainless steel CEA shrouds in Combustion Engineering plants and CRA guide tubes in Babcock & Wilcox plants.

Lower support columns and upper support columns in Westinghouse plants, and core support columns in Combustion Engineering plants are under compressive loads during normal operation. Furthermore, these components experience only coolant inlet temperatures, which

significantly reduces the potential for thermal aging. Cracking caused by thermal aging is unlikely, due to the absence of tensile stress, and the components are subject to limited crack growth because of the presence of compressive stress. Nevertheless, visual inspection conducted at periodic intervals are adequate to manage the effects of potential thermal aging for these support columns.

Lower support plates in Westinghouse plants are visually inspected at normal in-service inspection intervals. No cracking has been identified during such inspections that can be attributed to thermal aging. If flaws are detected in future inspections, ASME Code Section XI evaluation procedures can be used to determine the need for more frequent inspection or repair. Therefore, current inspection programs are adequate to manage thermal aging of cast austenitic stainless steel lower support plates in Westinghouse plants.

Therefore, visual (VT-3) inservice examination of the PWR reactor internals components fabricated from cast austenitic stainless steel, in accordance with the requirements for examination category B-N-3 of Subsection IWB, Section XI, is able to manage the effects of thermal aging, provided that such components are accessible, or can be rendered accessible by the removal of the core and/or other internals, for the examination.

A license renewal applicant intending to take credit for this effective program is responsible for the review/evaluation of plant-specific documentation, in order to assure that the program elements required to manage the effects of the potentially significant thermal aging, or the justified equivalent, are committed for use at their plant. If the elements of an adequate visual (VT-3) inservice examination program are not in place, or if the effected component is not accessible, or cannot be rendered accessible by the removal of the core and/or other internals, for the examination, then alternatives for managing the effects of thermal aging are provided in Section 6.

5.7 FATIGUE

Fatigue was identified as a potentially significant age-related degradation mechanism for some PWR reactor internals components in Section 4.10. Those fatigue-critical components are listed below.

W	CE	B&W
- Upper Support Plate	- CEA Shroud Bolts	- NONE (See Section 4.10)
- RCCA Guide Tubes	- Core Shroud Assembly	
- Upper Support Column Bolts	- Core Shroud Tie Rods	
- Core Barrel Nozzles	- Core Support Plate	
- Baffle / Former Assembly Bolts	- Core Support Columns	
- Upper Core Plate Alignment Pins		
- Lower Core Plate		
- Lower Support Columns		

Examination Category B-N-3 of Section XI, Subsection IWB, provides requirements for the visual (VT-3) examination of accessible surfaces of PWR core support structures that can be removed from the reactor vessel. These requirements refer to the relevant conditions defined in IWB-3510.2, which include "loose, missing, cracked, or fractured parts, bolting, or fasteners." Since manifestation of excessive fatigue damage is expected to be fatigue crack initiation on the surface of an affected item, the VT-3 examination is adequate for the detection of significant fatigue damage. If any relevant condition is identified, IWB-3142 provides options for the timely correction of the condition, such as: (*i*) acceptance by supplemental surface and/or volumetric examination, in order to characterize the indication more accurately; (*ii*) acceptance by analytical evaluation, which may include flaw evaluation and/or more frequent examination of the item; and (*iii*) acceptance by corrective measures, repairs, or replacement.

As an alternative to the inservice examination requirements of Examination Category B-N-3, the ASME Section III, Subsection NG design rules for fatigue can be used to demonstrate that the effects of potentially significant fatigue damage can be managed adequately. This alternative will vary, depending upon the fatigue design basis of the affected component.

5.7.1 Detailed Fatigue Analysis

For those fatigue-critical PWR internals components without an explicit fatigue design basis, fatigue usage factors can be calculated, in accordance with the ASME Code, Section III, Subsection NG. These procedures include the exemption from detailed fatigue analysis for PWR internals, if their specified service loads satisfy certain limitations on severity. The types of loads considered are:

- Temperature Difference - Startup and Shutdown
- Temperature Difference - Normal Service
- Temperature Difference - Dissimilar Materials
- Mechanical Loads

The criteria which must be satisfied for fatigue analysis exemption involve relationships between the service loads, allowable stress intensity, material properties, and the values of stress intensity obtained from the applicable design fatigue curves. For PWR internals, the thermal loads listed above are most pertinent. The exemption limits for thermal loads require that specified service temperature differences remain below calculated values based on the material's elastic moduli and thermal expansion coefficients, and the stress range obtained from the applicable point on the design fatigue curves. Service temperature differences are considered to be insignificant if they fall below these calculated values. If the exemption limits are not met in their entirety, detailed fatigue evaluation is required.

Detailed fatigue analysis involves the determination of a set of design transients associated with service loadings and their frequency of occurrence, and the calculation of the stresses and stress ranges at controlling component locations. These procedures have been described as they apply to components with an explicit fatigue design basis in Section 4.10 of this report. For ASME Code acceptance, the fatigue usage factor, as determined by these procedures, cannot exceed unity, as extrapolated through the design lifetime of the component.

5.7.2 Fatigue Re-Evaluation

For those fatigue-critical PWR internals components with an explicit fatigue design basis, detailed fatigue analysis using less conservative assumptions can be used to demonstrate that recalculated fatigue usage factors remain less than unity for the license renewal term. For example, such an approach has been used by Babcock & Wilcox on their PWR internals components [5-3]. These re-evaluations took advantage of inherent conservatisms in the ASME Code procedures and in the definition of the fatigue design basis transients. A key aspect of this re-evaluation was the use of a statistical distribution of stress amplitude peaks for some loading cycles, reflecting actual operating transient data [5-10]. Another consideration was the evaluation of high-cycle fatigue prior to the inclusion in the 1983 ASME Code of design fatigue curves for numbers of loading cycles beyond 10^6.

Prior to 1983 the material fatigue curves in the ASME Code were plotted out to one million (10^6) cycles. This was adequate for the evaluation of low-cycle fatigue, which derives primarily from steady-state and transient thermal conditions, but provided no guidance for evaluating the high-cycle fatigue damage that could result from flow-induced vibration. Vendor-specific extensions of the ASME Code curves were done conservatively. Therefore, re-evaluation of components whose high-cycle fatigue susceptibility is due to pre-1983 ASME material fatigue curves or to conservative extensions thereof may be of great benefit.

The conservatism of the fatigue design basis transients can be reduced by: (1) performing a review of operating records to determine historically significant transients; or (2) monitoring and recording actual operating transients.

5.7.2.1 Operational Data Review

A review of historical operating records can be undertaken to determine the actual number of design transients accumulated, classify partial cycles (reactor trips from 30 percent as opposed to full power, for example) and incorporate any transients that have occurred but were not considered in the design basis. This operating transient set can then be used as a basis for fatigue evaluation. Operational data reviews of this kind are described in [5-1, 5-3, 5-11, 5-12].

A simplification of this process can be achieved if it can be established that existing component fatigue evaluations are strongly influenced by certain transients which have occurred much less frequently or with substantially less severity than originally anticipated. Examples of this are postulated seismic events and load follow transients. The effects of seismic events and load follow transients which have not occurred during the design life can be eliminated from fatigue evaluations and appropriate projections made for the license renewal term. This kind of approach has been used in several studies [5-1, 5-3, 5-11, 5-12].

5.7.2.2 Transient Monitoring

A program to monitor and record data can also be used to provide information on operating transients. Such a program can be supplemental to an operational data review or can stand alone.

The final product of this process is a detailed transient data set on which realistic fatigue analyses of internals components can be based. Fatigue usage factors and, if necessary, fracture mechanics analyses can then be applied to the license renewal period with a minimum of excess conservatism.

5.7.3 Fatigue Conclusions

For those few cases when fatigue-critical components are essentially inaccessible to inservice examination, in accordance with Examination Category B-N-3, and where the cyclic loadings are sufficiently uncertain to preclude the effective use of detailed fatigue design analysis, alternatives for managing the effects of the age-related degradation are described in Section 6. Baffle-former assembly bolts in Westinghouse plants have been deemed to be in this category.

With this exception, visual (VT-3) examination of fatigue-critical PWR reactor internals components, in accordance with the requirements of Examination Category B-N-3 of Subsection IWB, Section XI, or detailed fatigue design analysis/reanalysis of fatigue-critical PWR reactor internals components, in accordance with the procedures of Section III, Subsection NG-5200, are able to manage the effects of potentially significant fatigue damage,

provided that such components are accessible for examination, or that the design basis transients for the fatigue analysis can be shown to be adequate.

A license renewal applicant intending to take credit for these effective programs is responsible for the review/evaluation of plant-specific documentation, in order to assure that the program elements required to manage the effects of the potentially significant fatigue, or the justified equivalent, are committed for use at their plant. If the elements of an adequate visual (VT-3) inservice examination program are not in place, or if the effected component is not accessible, or cannot be rendered accessible by the removal of the core and/or other internals, for the examination, or if the design basis transients for the fatigue analysis can not be shown to be adequate, then alternatives for managing the effects of fatigue are provided in Section 6.

5.8 EFFECTIVE PROGRAM ASSESSMENT SUMMARY

This section contains a summary of the Section 5 assessments of the effective programs available to address the potentially significant age-related degradation mechanism / component combinations identified in Section 4. The dispositions of all component / mechanism combinations evaluated in this report are given in Table 5.2. These dispositions include component / mechanism combinations identified in Section 4 as nonsignificant, component / mechanism combinations managed by the effective programs assessed in Section 5, and the component / mechanism combinations requiring plant-specific management (options provided in Section 6).

A license renewal applicant intending to take credit for these effective programs is responsible for the review/evaluation of plant-specific documentation, in order to assure that the program elements required to manage the effects of the potentially significant age-related degradation mechanisms under consideration, or the justified equivalent, are committed for use at their plant.

5.8.1 Irradiation Embrittlement

The effects of irradiation embrittlement as discussed in Section 5.1 are managed by ASME Section XI inservice examination procedures provided that the affected components are accessible for examination. If the elements of the ASME Section XI inservice examination program, or their justified equivalent, are not in place or if the affected component is not accessible, or cannot be rendered accessible by the removal of the core and/or other internals, for the examination, then alternatives for managing the effects of irradiation embrittlement are provided in Section 6.

5.8.2 Stress Corrosion Cracking

Stress corrosion cracking as discussed in Section 5.2 is managed by ASME Section XI inservice examination procedures provided that the affected components are accessible for examination. If the elements of the ASME Section XI inservice examination program, or their justified equivalent, are not in place or if the affected component is not accessible, or cannot

be rendered accessible by the removal of the core and/or other internals, for the examination, then alternatives for managing the effects of stress corrosion cracking are provided in Section 6.

5.8.3 Irradiation-assisted Stress Corrosion Cracking

Irradiation-assisted stress corrosion cracking as discussed in Section 5.3 is managed by ASME Section XI inservice examination procedures for accessible PWR RPV internals components. If the elements of the ASME Section XI inservice examination program, or their justified equivalent, are not in place or if the affected component is not accessible, or cannot be rendered accessible by the removal of the core and/or other internals, for the examination, then alternatives for managing the effects of irradiation-assisted stress corrosion cracking are provided in Section 6.

5.8.4 Stress Relaxation

Except for baffle / former assembly bolts in Westinghouse and B&W plants and core shroud assembly bolts in CE plants, the effects of stress relaxation as discussed in Section 5.4 are managed by ASME Section XI inservice examination procedures for those PWR reactor internals components requiring adequate preload to perform their intended function provided that the affected components are accessible for examination. If the elements of the ASME Section XI inservice examination program, or their justified equivalent, are not in place or if the affected component is not accessible, or cannot be rendered accessible by the removal of the core and/or other internals, for the examination, then alternatives for managing the effects of irradiation embrittlement are provided in Section 6. Options for managing the effects of stress relaxation on the aforementioned exceptions are provided in Section 6.

5.8.5 Wear

As delineated in Section 5.5, the effects of wear are adequately managed by ASME Section XI inservice examination procedures, including visual examination (VT-3) and Code-defined alternate NDE methods.

5.8.6 Thermal Aging

The effects of thermal aging of cast austenitic stainless steel PWR RPV internals components as discussed in Section 5.6 are managed by ASME Section XI inservice examination procedures provided that the affected components are accessible for examination. If the elements of the ASME Section XI inservice examination program, or their justified equivalent, are not in place or if the affected component is not accessible, or cannot be rendered accessible by the removal of the core and/or other internals, for the examination, then alternatives for managing the effects of irradiation embrittlement are provided in Section 6.

5.8.7 Fatigue

As described in Section 5.7, except for baffle / former assembly bolts in Westinghouse plants, fatigue is managed for those components deemed to be fatigue-critical either by ASME Section XI inservice examination procedures or by detailed fatigue design analysis / reanalysis in accordance with ASME Section III Subsection NG procedures. If the elements of the ASME Section XI inservice examination program, or their justified equivalent, are not in place or if the affected component is not accessible, or cannot be rendered accessible by the removal of the core and/or other internals, for the examination, or if the design basis transients for the fatigue analysis can not be shown to be adequate, then alternatives for managing the effects of irradiation embrittlement are provided in Section 6. Options for management of fatigue effects on Westinghouse baffle / former assembly bolts is provided in Section 6.

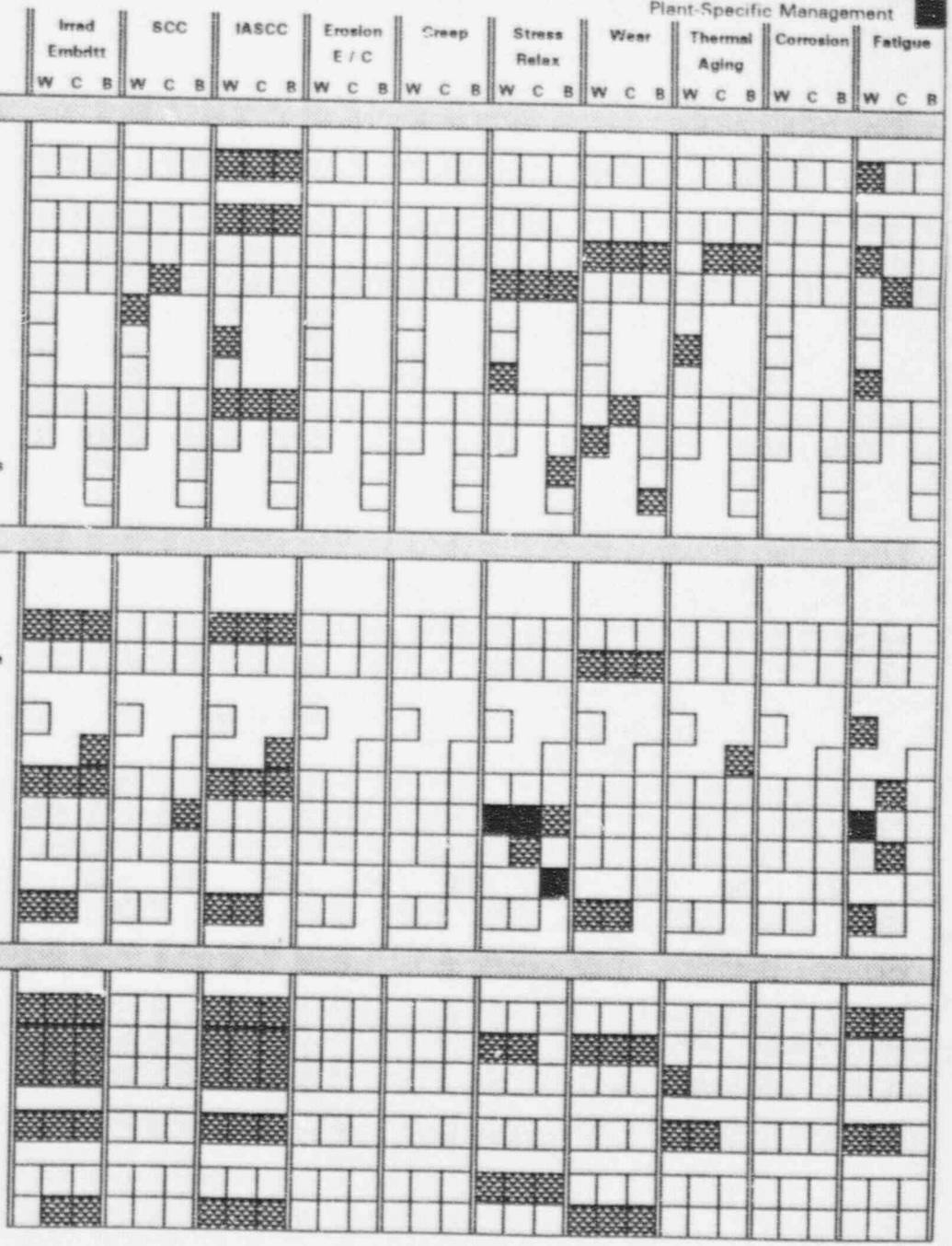
Table 5.2

COMPONENT / MECHANISM DISPOSITION SUMMARY

for each STRUCTURAL ASSEMBLY GROUPING

Key: No Significant Degradation
Effective Program
Plant-Specific Management

WESTINGHOUSE	COMBUSTION ENGINEERING	BABCOCK & WILCOX
UPPER INTERNALS ASSEMBLY		
Upper Internals Assembly	**Upper Guide Structure Assembly**	**Plenum Assembly**
1. Upper Support Plate	1. Upper Guide Structure Support Plate	1. Plenum Cover and Plenum Cylinder
2. RCCA Guide Tube Assemblies	2. CEA Shroud Assemblies	2. CRA Guide Tube Assemblies
RCCA Guide Tubes	CEA Shrouds	CRA Guide Tubes
RCCA Guide Tube Bolts	CEA Shroud Bolts	CRA Guide Tube Bolts
RCCA Guide Tube Support Pins		
3. Upper Support Columns	3. ----------	3. ----------
Upper Support Column Bolts		
4. Upper Core Plate	4. Fuel Alignment Plate	4. Upper Grid Assembly
Fuel Pins		Upper Grid Rib Section
		Upper Grid Assembly Bolts
		Fuel Guide Pads
CORE SUPPORT ASSEMBLY		
Core Barrel and Baffle/Former Assembly	**Core Support Barrel and Core Shroud Assembly**	**Core Support Assembly**
1. Core Barrel	1. Core Support Barrel	1. Core Support Shield
Upper Core Barrel Flange	Core Support Barrel Upper Flange	Core Support Shield Flange
Core Barrel Nozzles		
2. ----------	2. ----------	2. Vent Valve Assemblies
3. Baffle-Former Assembly	3. Core Shroud Assembly	3. Core Barrel Assembly
Baffle / Former Assembly Bolts	Core Shroud Assembly Bolts	Core Barrel Bolts
Baffle / Former Assembly Baffles	Core Shroud Tie Rods	Baffle / Former Plates
		Baffle / Former Bolts
4. Upper Core Plate Alignment Pins	4. Fuel Alignment Plate Guide Lugs	4. ----------
LOWER INTERNALS ASSEMBLY		
Lower Internals Assembly	**Core Support Assembly**	**Lower Grid Assembly**
1. Lower Core Plate	1. Core Support Plate	1. Lower Grid Top Rib Section
2. Fuel Pins	2. Fuel Alignment Pins	2. Fuel Guide Pads
3. Lower Support Plate	3. Lower Support Structure Beam Assemblies	3. Lower Grid Bottom Rib Weldment
4. Lower Support Columns	4. Core Support Columns	4. Lower Grid Assembly Support Posts
Lower Support Column Bolts	Core Support Column Bolts	Lower Grid Assembly Bolts
5. Radial Keys and Clevis Inserts	5. Core Support Barrel Snubber Assemblies	5. Lower Grid Cylinder Guide Blocks



5.8 SECTION 5 REFERENCES

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6.0 MANAGEMENT OPTIONS FOR POTENTIALLY SIGNIFICANT AGE-RELATED DEGRADATION

In the previous section of this report, effective programs of inservice inspection, testing, surveillance, and analytical assessment were evaluated with respect to their capability to manage the effects of potentially significant age-related degradation. If the elements of an effective program are determined to be capable of managing the effects of potentially significant age-related degradation for a particular component so that its intended safety function was not compromised during the license renewal term, then that combination of component and degradation mechanism was deemed to be adequately addressed.

A license renewal applicant intending to take credit for the effective program is responsible for the review/evaluation of their related plant-specific features, including appropriate CLB documents/information, in order to assure that the program elements used to manage the effects of potentially significant age-related degradation, or their equivalent, are committed for use at their plant.

Age-related degradation/component issues for which effective program elements cannot be shown to adequately manage the effects of potentially significant age-related degradation, or for which license renewal applicants are unable to demonstrate that the stipulated program elements are in place, require plant-specific evaluation.

On the basis of the evaluations provided in Sections 4 and 5 of this report, numerous component-degradation mechanism combinations were resolved relative to license renewal considerations. The component-degradation mechanism combinations were determined to be not significant because: (1) either the component is not susceptible to the age-related degradation mechanism under consideration, or is susceptible to such a small degree that the component's safety function would be maintained throughout the license renewal term; or (2) established regulations, tests, inspections, and analytical procedures (acceptable programs) are able to manage potentially significant age-related degradation such that the component's safety function would be maintained throughout the license renewal term.

6.1 Assessment of Significance

From the conclusions of the Sections 4 and 5 evaluations, which are summarized in Table 5.2, two specific age-related degradation/component combinations were identified in which effective program elements were found to be inadequate to manage the effects of the potentially significant age-related degradation. These combinations are (1) stress relaxation of one set of bolts in the Core Support Assembly component grouping (specifically, Westinghouse baffle/former assembly bolts, CE core shroud assembly bolts and B&W baffle/former assembly bolts) and (2) fatigue of Westinghouse baffle/former assembly bolts. Plant-specific options exist to manage both of these issues. These options include taking credit for redundancy in these multiple structure support systems, evaluation of actual/postulated

flaws in these components and applying ultrasonic testing techniques to determine the presence of flaws in these bolts. More detailed discussion of these options is provided in Section 6.2.

Additionally, for those individual cases where a utility determines that the effective program elements outlined in Section 5 are not fully implementable or where a utility may judge that the effective program elements are not fully effective for that utility's specific design, then alternatives to these program elements may also be available. A few of these alternatives are covered by the aging management options presented in Section 6.2.

6.2 Aging Management Options

6.2.1 Mechanical Redundancy

A number of PWR reactor internals assemblies rely on multiple structural support systems. The most obvious examples of a multiple structural support system in the PWR reactor internals is bolted connections. Note, in particular, that the Westinghouse and B&W baffle/former assembly bolts and the CE core shroud assembly bolts are examples of this type of multiple structural support system. Potentially significant degradation mechanisms -- irradiation embrittlement, stress corrosion cracking, irradiation-assisted stress corrosion cracking, stress relaxation, and fatigue -- which were identified in Section 5 as possibly affecting these types of mechanically redundant components can be shown to be managed by mechanical redundancy.

Analytical demonstrations show that failure of a limited number of such structures does not necessarily imply failure of the assembly as a whole, i.e., the multiplicity of support structures provides mechanical redundancy. Analyses to demonstrate mechanical redundancy have been performed in numerous PWR applications. Evaluations were performed to show that the implications of the observed thermal shield bolt and guide block failures at Oconee 1 [6-1] did not pose a potential safety question. Parametric studies have been made to determine the acceptable number of failed bolts in the CEA shroud-fuel alignment plate, core shroud-guide lug, and guide lug-fuel alignment plate joints at Calvert Cliffs [6-2]. An analysis to determine the minimum number of support blocks required to maintain fluid-elastic stability of a thermal shield has also been performed [6-3].

EPRI Report NP-5769 [6-7] presents the results of a study which assesses the integrity of bolted closure assemblies which are part of the primary pressure boundary of pressurized water reactors. The study examines the effects of contiguous bolt failures in steam generator manway, reactor coolant pump, and check valve bolted closure assemblies. While the explicit purpose of the study was to determine closure acceptance through a leak-before-break criterion, the results suggest that load redistributions to the nondegraded bolts do not necessarily lead to additional bolt failures.

Mechanical redundancy considerations can be used to demonstrate acceptable performance of multiply supported internals assemblies during the license renewal term.

6.2.2 Evaluation of Actual / Postulated Flaws

Evaluation of actual (detected) or postulated (reference) flaws is an accepted technique for justifying continued operation of nuclear power plant components. Such methodology may also be useful for evaluation of components, such as baffle/former assembly bolts in Westinghouse plants, where current inservice examination or design analysis procedures cannot be shown to manage the effects of potentially significant fatigue damage adequately. Potentially significant degradation mechanisms -- irradiation embrittlement, stress corrosion cracking, irradiation-assisted stress corrosion cracking, thermal aging and fatigue -- which were identified in Section 5 as possibly affecting a number of the components covered by this report can be shown to be managed using flaw evaluation techniques.

For example, no specific guidance is provided in Subsection IWB or the Section XI appendices for justifying continued operation of a PWR internals component with a flaw by analytical evaluation, in accordance with IWB-3142.4. However, Appendices A (Analysis of Flaws), C (Evaluation of Flaws in Austenitic Piping), and H (Evaluation of Flaws in Ferritic Piping) do provide fracture mechanics methods that can be adapted for PWR internals component assessments. These appendices include systematic procedures for flaw model selection, calculation of stress intensity factors, material property determination, and design transients to be considered in component analysis models. The elastic-plastic fracture mechanics techniques described in Appendix C, in particular, provide a suitable framework for the evaluation of flaws and the justification for continued operation with flaws for PWR RPV internals.

Fracture mechanics techniques can be employed for justification of continued operation in two ways. First, inspection results may identify flaws that can be shown by analysis either not to grow as the consequence of continued service, or to grow at such a low rate that current inservice inspection schedules (or more frequent inservice inspections) will be able to assure the integrity of the component. Second, flaws can be postulated in the component and shown either not to grow or to grow very slowly. Through such a "flaw tolerance" approach, appropriate inservice inspection schedules can be formulated and compared to current requirements. This comparison may show that the current schedule is adequate, may need to be accelerated, or can be less frequent. Also, an appropriate schedule can be determined for components not included in the current inspection program.

In particular, an alternative approach for fatigue-critical PWR internals components, irrespective of the existence of an explicit fatigue design basis, could be the justification of continued operation of the component in the presence of either a real (detected) or an assumed (reference) flaw. For the case of a relevant condition involving a crack-like indication (see IWB-3520.2(b)), where supplementary examination in accordance with IWB-3142.2 has characterized an unacceptable flaw, the justification for continued operation involves a crack growth assessment. For the case where the need for, or the adequacy of, current or augmented inservice inspection coverage is sought, a reference flaw may be postulated and the growth of such flaws determined under plant operating conditions. In either case, an analysis of fatigue crack propagation is an essential feature of the evaluation process.

Section XI of the ASME Code provides systematic procedures for flaw model selection and evaluation using fracture mechanics analysis. While there are a number of fracture mechanics models available, the most common formulation is of the form:

$$da/dN = A\ (\Delta K)^n$$

where da/dN is the fatigue crack growth rate (in./cycle), ΔK is the stress intensity factor range in ksi-in.$^{0.5}$ and A and n are constants usually dependent on material and environment. For example, in Section XI of the ASME Code, the fatigue crack growth behavior for stainless steel in air is shown to be well described by:

$$da/dN = C_o\ (\Delta K)^{3.3}$$

where C_o is a scaling parameter which accounts for temperature and R ratio. The R ratio is defined as the ratio of minimum and maximum stress intensities during the loading cycle.

While estimates of the influence of PWR and BWR water environment effects on stainless steel base and weld metal have been made, ranging from a factor of 2 to 10 higher in crack growth rate than rates in air, the values used in license renewal applications should be justified on a plant-specific basis. For most situations, fatigue crack growth rates in PWR water environments are expected to be minimal. Similarly, it has been demonstrated [6-4] that fast neutron irradiation has no significant effect on the fatigue crack growth of austenitic stainless steels for fluence levels of interest here. Further, the work of [6-5] and [6-6] has demonstrated that C_o is not substantially impacted by the PWR water environment.

6.2.3 Ultrasonic Testing

Ultrasonic Testing (UT) is a nondestructive examination technique which has successfully been used to detect flaws in fasteners and other components. Ultrasonic testing is useful for the evaluation of components such as baffle/former assembly bolts in Westinghouse plants where detection of flaws is an essential part of bolt aging management. Potentially significant degradation mechanisms -- irradiation embrittlement, stress corrosion cracking, irradiation-assisted stress corrosion cracking, thermal aging and fatigue -- which were identified in Section 5 as possibly affecting a number of components covered by this report can be shown to be managed using UT. For bolt stress relaxation, a potential also exists for applying UT inspection techniques given appropriate baseline data on as-built bolt dimensions. The utility engineer should realize, however, that this baseline data does not commonly exist.

UT techniques, like the Cylindrically Guided Wave Technique used to detect flaws in bolts and threaded rods, use transducers which emit ultrasonic waves that travel through solids and liquids at different velocities. Based on these velocity differences, UT devices and signal processing equipment identify the presence of fluid-filled gaps (i.e., component flaws) in the ultrasonic wave pathways. Once developed and calibrated appropriately, UT examination can be an accurate and reliable inspection method for detecting flaws at critical locations such as

the threaded region of bolts. UT examination techniques for internals bolts must be customized for specific geometrical configurations, the presence of welded attachments to the fastener heads, and/or accessibility restrictions.

Description of conventional and more recently developed UT bolt inspection techniques are provided in [6-7]. A UT technique which has been successfully used to inspect steam generator support bolts, and which may be advantageous for some bolt configurations, is discussed in [6-8]. Finally, the use of UT techniques to detect flaws in the baffle-former assembly bolts of French PWRs is described in [6-9].

6.3 SECTION 6 REFERENCES

[6-1] S. M. Stoller Corporation, Nuclear Power Experience, January 1973 to January 1989.

[6-2] CEN-282, "Investigation and Evaluation of A286 Bolt Application in C-E's Nuclear Steam Supply System," Combustion Engineering, Inc., October 1984.

[6-3] Schwirian, R. E., D. R. Bhandari and S. Chandra, "Stability of the NUSCO Nuclear Reactor Thermal Shield With Assumed Degraded Support Conditions," Fluid Transients in Fluid-Structure Interaction, ASME FED-Vol. 56, pp. 79-86, December 1987.

[6-4] James, L. A., "Fatigue-Crack Propagation in Austenitic Stainless Steels," Atomic Energy Review, Volume 14, No. 1, 1976, pp. 37-86.

[6-5] James, L. A. and D. P. Jones, "Fatigue Crack Growth Correlation for Austenitic Stainless Steel in Air," in Predictive Capabilities in Environmentally Assisted Cracking, ASME Publication PVP-99, December 1985.

[6-6] Bamford, W. H., "Fatigue Crack Growth of Stainless Steel Reactor Coolant Piping in a Pressurized Water Reactor Environment," ASME Trans. Journal of Pressure Vessel Technology, February 1979.

[6-7] Nickell, R. E., "Degradation and Failure of Bolting in Nuclear Power Plants," EPRI Report NP-5769, Volume 1, Section 3, April 1988.

[6-8] Jusino, A., "In Situ Ultrasonic Examination of High Strength Steam Generator Support Bolts," presented at the 12th Biennial Conference on Reactor Operating Experience - 'Maintenance and Modifications for Availability and Efficiency'," (ANS Transactions), Williamsburg, VA, August 4-7, 1985.

[6-9] Bodson, F., A. Martin, and A. Thomas, "Recent Developments In the Inspection of Primary System Equipment. Measurements Associated With Mechanical Component Replacement," FRAMATOME 3N89, October 18, 1989.

PWR INTERNALS IR: RESPONSE TO NRC COMMENTS

NRC COMMENT #	NRC COMMENT	INDUSTRY RESPONSE	STATUS	PROPOSED IR REVISION	REMARKS
G-1	The arguments for license renewal presented in the IR are based in large part on the existence of on-site inspection, maintenance and replacement programs. However, existence of these programs does not necessarily provide adequate grounds for life extension. The Individual Plant Assessment (IPA) includes an evaluation of programs to determine effectiveness.	The evaluation of potentially significant age-related degradation mechanisms in Section 5 of the IR depends on the determination that the technical elements of the aging management programs when implemented can manage the age-related degradation. A license renewal applicant wishing to take credit for a generic conclusion in Section 5 is responsible for assuring that the referenced aging management program elements are in place. The NRC is responsible for enforcement and audit functions. It is the combination of IR findings, applicant affirmations,and NRC audit/enforcement that permits the applicant's license to be renewed. There is no conflict among the three.	Closed	Standardized language revisions to the IR, particularly in Section 5, will provide greater clarity in the definition of IR determinations and applicant responsibilities.	General Remark Throughout this Comment Resolution Document the NRC Comments reference the page and section numbers in the September 1990 draft of this IR, while the Proposed IR Revisions reference the revised report page and section numbers. It is believed that the response to this NRC comment fully resolves the concerns raised by the NRC. NUMARC requests that the NRC consider this comment "closed" based upon the response.

PWR INTERNALS IR: RESPONSE TO NRC COMMENTS

NRC COMMENT #	NRC COMMENT	INDUSTRY RESPONSE	STATUS	PROPOSED IR REVISION	REMARKS
G-2	The discussion for many of the degradation mechanisms concludes that an applicant seeking license renewal need not evaluate these mechanisms further, beyond assurance by the license renewal applicant that no plant-specific design and operating features exist which would preclude the applicant from verifying these conclusions. It is not clear from these statements what the IR requires the applicant to do. How can one verify the conclusions without performing an evaluation? The IR should provide more specific instructions to the users and clearly define the criteria for screening plant specific features. Otherwise, the evaluations presented in it will have little value.	In accordance with agreements reached between NRC staff and NUMARC representatives on July 24, 1990, the common sentence referred to in the comment will be replaced by different language that provides more specific guidance to license renewal applicants regarding the review of plant-unique design features, construction records, and operating and maintenance history.	Closed	Section 4 of the report will be modified with the following text: License renewal applicants to reference these generic conclusions are responsible for a review/evaluation of specific features, including appropriate current licensing basis (CLB) documents/information, in order to assure that there are no deviations from the assumptions and criteria used in the report. This review should compare the design basis for particular components with the representative design bases given in Sections 3.1 and 3.2. The component operation and maintenance histories should also be compared to the generic performance parameters described in Section 3.3. Finally, the specific assumptions and criteria used in this section should be examined to assure that they, or justified equivalents, apply to the component under consideration. [p. 4-1, Para. 4]	It is believed that the response to this NRC comment fully resolves the concerns raised by the NRC. NUMARC requests that the NRC consider this comment "closed" based upon the response. This comment is similar to comments received for other IRs, for example comment G-2 from the BWR containment IR. The same response is provided for which Agreement-in-Principle was reached from the February 20, 1991 meeting with the NRC.

PWR INTERNALS IR: RESPONSE TO NRC COMMENTS

NRC COMMENT #	NRC COMMENT	INDUSTRY RESPONSE	STATUS	PROPOSED IR REVISION	REMARKS
G-3	The IR, as written, does not clearly specify the assumptions that were made in performing the degradation analysis presented in it. Since the IR is expected to be widely used for license renewal purposes, the report should include a complete listing of all assumed parameters and, preferably, a user-oriented checklist such that the user can verify the applicability of the assumptions that were made in performing the degradation analyses.	We have agreed to provide clear criteria in all of the IRs that license renewal applicants can use to determine the applicability of the technical conclusions reached in Section 4 and 5. These criteria will take the form of "if.....then" statements, where possible, that spell out the requirements to be met. We believe that this IR contains the criteria needed, but we agree that the criteria will be reviewed for clarity of presentation. Specific cases of unclear criteria are addressed later in the specific comment responses.	Closed	Changes to the report, as necessary, to clarify criteria, using NUMARC standard language.	It is believed that the response to this NRC comment fully resolves the concerns raised by the NRC. NUMARC requests that the NRC consider this comment "closed" based upon the response.
G-4	A large number of components are dismissed in the IR from consideration for license renewal based on the arguments that these components do not perform any safety functions. While these components may not have direct safety functions to play in a normal operating condition or a design basis accident, their failure could jeopardize monitoring of the reactor or produce loose parts that could prevent safety related components from functioning properly. If failure of these components could prevent safety-related components from functioning properly, are the components important to license renewal? The IR should address this issue by either demonstrating these components are not important to license renewal or providing the technical basis for their dismissal.	The IR will be revised to eliminate the valuation of component safety significance now contained in Section 3. Instead, the report will identify clearly the internals components that are within its scope and those that are outside its scope. All components that are within the scope of the report will be described in Section 3.1, have its design bases described in Section 3.2, any relevant operating history described in Section 3.3, and be evaluated with respect to age-related degradation in Section 4. No determination will be made with respect to whether the components are important to license renewal. Such determinations are the responsibility of license renewal applicants.	Closed	Substantial revisions will be necessary for Sections 2 and 3.	It is believed that the response to this NRC comment fully resolves the concerns raised by the NRC. NUMARC requests that the NRC consider this comment "closed" based upon the response.

PWR INTERNALS IR: RESPONSE TO NRC COMMENTS

NRC COMMENT #	NRC COMMENT	INDUSTRY RESPONSE	STATUS	PROPOSED IR REVISION	REMARKS
G-5	The terms wear and fretting were used somewhat loosely in the IR. Many of the components are stable in a reactor environment because of the presence of protective oxides. If these are removed by abrasion, rapid local oxidation (corrosion) can occur as the oxide films reform. This should be discussed in the IR, since it may play a role in the aging degradation phenomena.	The differentiation between "wear" and "fretting wear" as used in the IR is explained in Section 4.7.1, "wear" being a general term and "fretting wear" referring specifically to wear resulting from vibratory action. Both result from the relative motion of two components at their interface. Whether the material directly removed is the metal or its oxide is of secondary importance to the IR; the net effect is the same - a net loss of base material. However, Section 4.7.1. will be revised to reflect the fact that wear can result from either surface oxide removal or the direct removal of base material.	Closed	Section 4.7.I. will be rewritten as follows: Wear is defined as the removal of material surface layers due to relative motion between two surfaces or under the influence of hard, abrasive particles. Wear occurs in parts that experiencintermittent relative motion, in clamped joints where relative mion is not intended but may occur due to a loss of clamping force or as a result of flow-induced vibrations." [p. 4-19, Section 4.7.1] Wear, which is the result of the contact of two surfaces due to vibration or sliding (e.g., flow-induced vibration) while the surfaces are in the presence of a corrosive environment, is referred to as fretting wear. Another type of wear which may occur in PWRs and whic is not related to flow-induced vibration is that associated with the intentional displacement of adjacent components, such as the stepping of control rods. Wear can result from either surface oxide removal or the direct removal of base material.	It is believed that the response to this NRC comment fully resolves the concerns raised by the NRC. NUMARC requests that the NRC consider this comment "closed" based upon the response.

PWR INTERNALS IR: RESPONSE TO NRC COMMENTS

NRC COMMENT #	NRC COMMENT	INDUSTRY RESPONSE	STATUS	PROPOSED IR REVISION	REMARKS
G-6	Many combinations of significant degradation mechanisms and components have been dismissed from consideration for license renewal due to the reason that the respective NSSS vendor has recommended periodic inspection programs. Such recommendations alone are not adequate justification for license renewal. As a minimum, the IR should include guidance to be used by a licenses to answer the following questions: (a) Whether and how were these recommendations implemented? (b) How effective were the recommendations to correct situations? (c) Has the overall safety margin of the plant been compromised as a result of operating the plant in the degraded condition prior to implementation of the vendor's recommendations? (d) Since the extent and frequency of the recommended inspection were apparently based on the current licensing term, how effective will the same inspection be through the license renewal period given that the plant will be further aged?	The degradation mechanisms and internals components currently closed out based on vendor recommended, plant/component specific programs are SCC of selected Alloy X-750 and Alloy A-286 components and stress relaxation of preloaded bolts. In those cases involving SCC of Alloy X-750 and Alloy A-286 component replacement and inspections have been recommended. In those plants **which have replaced components, inspections show no additional SSC degradation has occurred. The affected components have not compromised the overall plant safety margins of the plant.** **For cases involving stress relaxation of preloaded bolts, the mechanical redundancy argument can be used on a plant specific basis (Section 6) to demonstrate that degradation will not cause loss of function of the system that compromise the overall safety margin of the plant.** Moreover, it is recognized that research is continuing to better characterize and inspect for relaxation of preloaded bolts. Quantitive inspections for stress relaxation are not currently in use; therefore, the IR will be modified such that stress relaxation of preloaded bolts is placed in Section 6. It should be recognized, however, that inspections to detect stress relaxation will in most cases be geared toward detection of the effects of stress relaxation (e.g. - fret marks, loosening, etc.) rather than detection of the mechanism itself. In a limited number of cases, sufficient records of initial preload, bolt length, etc. may be available that a direct assessment of stress relaxation is possible.	Closed	Section 5.2 of the IR will be modified to state, the following: A license renewal applicant is responsible for reviewing their related plant-specific documentation, in order to assure that the vendor plant specific elements required to manage the effects of SCC on Alloy X-750 and Alloy A-286 components are committed for use at their plant. [p. 5-6, Para. 2] Section 6.0 of the IR will be modified to include stress relaxation of preloaded bolts which cannot be inspected by current and effective methods. Aging management strategies including inspection or mechanical redundancy assessment will be recommended. [p. 6-2, Section 6.2.1]	It is believed that the response to this NRC comment fully resolves the concerns raised by the NRC. NUMARC requests that the NRC consider this comment "closed" based upon the response. It is believed that the response to this NRC comment fully resolves the concerns raised by the NRC. NUMARC requests that the NRC consider this comment "closed" based upon the response.

NRC COMMENT #	NRC COMMENT	INDUSTRY RESPONSE	STATUS	PROPOSED IR REVISION	REMARKS
G-7	The degradation of certain components (e.g., support columns, bolts and pins) was considered acceptable in the IR since these components represent mechanically redundant systems. For example, it is argued that failure of a few bolts is acceptable since that may not constitute failure of the bolt assembly. Such qualitative arguments are inadequate to make a case for license renewal since it is customary in all engineering designs, especially in the nuclear industry, to provide some amount of redundancy and margin. As a minimum, the following should be addressed: (a) Be specific about the number of components (e.g., bolts) that are being sacrificed. (b) Provide calculations to support acceptance of failure of the redundant components. (c) Quantify the safety margin that is being sacrificed and justify the adequacy of the remaining safety margin. (d) It was stated in the IR that for some components a periodic inspection will be adequate for detection of a progressive failure of the system. Provide discussions as to how the frequency of inspection was determined to guarantee a timely detection.	Because of the variation in reactor internals configurations, a generic quantitative analysis to demonstrate the efficacy of mechanical redundancy arguments would be an extensive effort. Practical considerations preclude this type of extensive undertaking. Hence, mechanical redundancy will not be used in Section 4 or 5 of the IR for closing issues. Mechanical redundancy assessments will be listed in Section 6 of the IR as one potential plant-specific aging management strategy. This action closes out all parts of comment G-7 (a) - (f).	Closed	Revise IR to delete credit in Sections 4 and 5 for mechanical redundancy considerations as a means of closing issues. However, it will be noted in Chapter 6 that analysis based on mechanical redundancy is an approach that license renewal applicants may wish to consider on a plant-specific basis. [p. 6-2, Section 6.2.1]	It is believed that the response to this NRC comment fully resolves the concerns raised by the NRC. NUMARC requests that the NRC consider this comment "closed" based upon the response.

NRC COMMENT #	NRC COMMENT	INDUSTRY RESPONSE	STATUS	PROPOSED IR REVISION	REMARKS
	(e) Since all of the subject components are being equally aged and many of them are probably equally stressed, what is the rationale for not considering a simultaneous failure of a large number of such components resulting in an overall catastrophic failure of the system? The following two additional factors should also be considered in the evaluation: 1. Failure of some components will result in redistribution of loads and, probably, higher stresses in the remaining components. 2. Failure of any component may cause impact resulting in instantaneous higher stresses in the remaining components. (f) Failure of certain components in a system should be evaluated on a case-by-case basis. Therefore, it is not clear how the IR proposes acceptance of such failures on a generic basis resulting in dismissal of certain degradation mechanisms and internal components from consideration for license renewal. If the current licensing basis cannot be maintained for these components, this should be clearly stated and guidance should be provided to licensees to evaluate this issue.				

PWR INTERNALS IR: RESPONSE TO NRC COMMENTS

NRC COMMENT #	NRC COMMENT	INDUSTRY RESPONSE	STATUS	PROPOSED IR REVISION	REMARKS
G-8	The IR does not include a discussion on the seismic and environmental qualification of the internals. The IR should demonstrate continued qualification of the components considering the degradation that will occur during the license renewal period. Both the structural integrity and the functional operability of the degraded components should be demonstrated under the design basis events (e.g., SSE and LOCA).	Since the yield strength of the materials of construction increases with increasing "age" due to irradiation and since the structural integrity of the component under design basis events has been assured by meeting stress limits, there would be an improvement in margin with aging. However, some components were also designed based on deformation limits. Because of potential embrittlement due to irradiation and thermal aging, the acceptability of the expected deformations should be confirmed, particularly for plants that have not converted to "leak-before-break" design basis.	Closed	The IR will be revised to recommend in Section 2.6 that license renewal applicants confirm the acceptability of functional deformation limits under design basis events. [p. 2-5 Para. 3]	It is believed that the response to this NRC comment fully resolves the concerns raised by the NRC. NUMARC requests that the NRC consider this comment "closed" based upon the response.
G-9	Since many of the internal components exhibit cracking the IR recommends the use of fracture mechanics evaluation as a tool for license renewal. However, since inspection of the inner components, supports and attachment welds are difficult due to access limitation, it is not clear how a fracture mechanics evaluation can be reliably performed in addressing the significant degradation mechanisms such as fatigue, neutron embrittlement and IGSCC. A fracture mechanics analysis can, at most predict the critical size and the growth rate of a given crack (detected or postulated), provided appropriate crack growth models are available. For this reason, in spec-tion should be performed to ensure that the actual crack growth rate is bounded by the predicted model.	We cannot agree that many of the internals components exhibit cracking since this has not been the experience with PWR internals. We agree with the reviewers comments on the need to couple the use of fracture mechanics to inspections or conservative assumptions. Background is presented in the following paragraphs: When aging degradation is identified for specific reactor internal components, fracture mechanics can be used as a generic tool in evaluation of flaws observed by inspection or from hypothesized assumptions. Fracture mechanics is a methodology for assessing the potential for crack extension by fatigue and ductile tearing and for crack instability. For the stainless	Closed	Because fracture mechanics analysis methods for PWR internals to date have been performed on a plant specific basis, all fracture mechanics discussions will be revised and moved to Section 6. Components which cannot be inspected by the current and effective procedures identified in Section 5 will be listed. Provisions for inspecting such compon-ents must be made or component aging manage-ment would be addressed in Section 6. [pp. 6-3 and 6-4, Sections 6.2.2 and 6.2.3]	It is believed that the response to this NRC comment fully resolves the concerns raised by the NRC. NUMARC requests that the NRC consider this comment "closed" based upon the response.

NRC COMMENT #	NRC COMMENT	INDUSTRY RESPONSE	STATUS	PROPOSED IR REVISION	REMARKS
	In addition, inspection is required to ascertain that the flaw does not reach a critical size that has been established by the fracture mechanics evaluation. The IR should clearly indicate how the fracture mechanics evaluation will be utilized as a tool for license renewal for inaccessible components.	steel internals under service conditions, crack instability is expected to be characterized by ductile tearing not brittle fracture. Elastic plastic fracture mechanics will most likely be required for fracture assessments. Applications may be made both to actual situations where cracks have been observed and to situations where the postulation of cracks is prudent, such as in inaccessible components or portions thereof. It can then be used in conjunction with inspection. For reactor vessel internal structural components, both fracture toughness (J_{IC} and J-R response) and fatigue crack growth rates (da/dn vs delta K) are required to make meaningful not-overly conservative evaluations of actual or postulated cracks. Research of irradiation effects on stainless steels has been in progress for over two decades and is continuing. Thermal aging of cast austenitic stainless steels has been investigated for a comparable length of time. Fatigue crack growth does not appear to be significantly affected by irradiation and thermal aging (G-9A). An environmental factor yet to be specified for the fatigue crack growth law of Section XI of the			

PWR INTERNALS IR: RESPONSE TO NRC COMMENTS

NRC COMMENT #	NRC COMMENT	INDUSTRY RESPONSE	STATUS	PROPOSED IR REVISION	REMARKS
		ASME Code is expected to account for irradiation and thermal aging based on available data and is under development. It is reasonable to expect that fluence-toughness-at-temperature mechanistic models will be available at the time of need. Time-temperature-toughness behaviors of the cast material response to thermal aging should also be in hand. However, it should be kept in perspective that even a forty-year life extension is not expected to degrade the material thoughness significantly beyond that of the initial forty years of service. The exponential characteristics of thoughness degradation from irradiation of austenitic stainless steels (also that of the cast material for thermal aging) clearly indicate this. For example the fracture toughness for a fluence of $1X10^{21}$ or $2 X 10^{21}$ n/cm^2 usually falls within the scatter-band of the data. For cracks observed by inspection in internal components, it is reasonable to assume that either viable toughness and fatigue-crack growth models are available or will be available in a timely manner for license renewal considerations. For observed cracks, integrity evaluation should be rather straight forward.			

NRC COMMENT #	NRC COMMENT	INDUSTRY RESPONSE	STATUS	PROPOSED IR REVISION	REMARKS
		For inaccessible components, engineering judgment must be made in postulating potential flaws. Convincing engineering judgments or postulated flaws can be based on material and material fabrication history, stress analysis results, response for similar application and general field experience. Traditionally the quarter thickness flaw approach has found favor with the ASME Code Committee and with regulatory bodies since they often have representation on the Code Committee and also often are called on to approve such code application in the licensing and safety arena. For an inspectable component (the reactor vessel), an example can be found in Appendix G of Section III of the ASME code. In ASME Code Case N-481, a fracture mechanics integrity analysis based on a quarter thickness flaw actually replaces volumetric inservice inspection. As a provisional flaw size in an inaccessible component, a quarter thickness flaw(conceptionalized to a specific geometry) is recommended. With such a			

NRC COMMENT #	NRC COMMENT	INDUSTRY RESPONSE	STATUS	PROPOSED IR REVISION	REMARKS
		postulated flaw size, a crack growth and crack stability analysis may be performed for inaccessible components. An alternate approach for inaccessible components would be to establish a critical flaw size by elastic plastic fracture mechanics and then determine the flaw size for the beginning of life extension. Subsequently postulate growth by fatigue to critical size at the end of the life extension. This would form a basis for conservatively estimating a tolerable flaw size for judging the likelihood of such a flaw existing in inaccessible components. REFERENCE: (G9-A) L.A. James,: Fatigue Crack Propagation in Austen tic Stainless Steels," Atomic Energy Review, Vo.. 14, No. 1. pp 37-86, 1976.			
G-10	The IR did not address creep or the effect of high temperature as a degradation mechanism that can lead to failure of internal components. For example, the core plates (top and bottom) can undergo large deformation due to high temperature and excessive bending stresses due to "bowing" of the plates and reduction of the opening sizes. The IR should examine the effect of creep and high temperature as mechanisms for license renewal purposes.	The IR does address the phenomenon of creep degradation in Section 4.5. Creep is not a concern for the types of alloys used for the PWR internal components at temperatures below 1000^{0}F. The comment regarding large deformations due to high temperature is confusing. There is no known occurrence of "bowing" of the core plates.	Closed	None	It is believed that the response to this NRC comment fully resolves the concerns raised by the NRC. NUMARC requests that the NRC consider this comment "closed" based upon the response.

PWR INTERNALS IR: RESPONSE TO NRC COMMENTS

NRC COMMENT #	NRC COMMENT	INDUSTRY RESPONSE	STATUS	PROPOSED IR REVISION	REMARKS
G-11	A description of the weld materials and processes and materials for other types of attachments is not available in the report. The IR should identify the materials and the failure mechanisms of the attachment of the internal components, and demonstrate their acceptability for license renewal. A table listing the materials used for each component and its attachment is recommended.	The PWR internals IR is intended to address three vendors and their specific types of internals. A complete list of specific weld materials and processes is not within the scope of the IR. However, the IR will be modified to include a table which provides categories of materials and examples of each.	Closed	A table will be added to Section 3.0 of the IR to address categories of materials, including welds, and examples of each. [p. 3-9, Table 3.3]	It is believed that the response to this NRC comment fully resolves the concerns raised by the NRC. NUMARC requests that the NRC consider this comment "closed" based upon the response. This comment is similar to comment G-22.
G-12	The effect of life extension or any existing safety assessment study should be considered for license renewal. This should be addressed in the IR.	The evaluation of existing safety assessments, would be for the purpose of identifying systems, structures, and components (SCC) important to license renewal. The approach of the Industry Reports (IR) is not to make this determination. It is not the function of the IR to determine whether or not a particular SSC is important to license renewal. It is the responsibility of the individual applicant in accordance with the final rule to determine which SSC for their plant are important to license renewal. If an SSC which is important to license renewal and has been explicitly evaluated in an IR and that has been approved by the NRC, then the applicant may choose to use the IR to justify their individual plant assessment (IPA). It would be inappropriate for an IR to attempt to supersede the regulatory authority of the License Renewal Rule by trying to predetermine which SSCs will be important to license renewal.	Open	Incorporate the revised Standardized Language in the IR.	It is believed that the response to this NRC comment fully resolves the concerns raised by the NRC. NUMARC requests that the NRC consider this comment "closed" based upon the response.

NRC COMMENT #	NRC COMMENT	INDUSTRY RESPONSE	STATUS	PROPOSED IR REVISION	REMARKS
		It is recognized in the IRs that an applicant needs to consider plant specific operating history. A plant specific operating history review may include, but is not limited to, safety assessments when evaluating whether or not the age-related degradation conclusions presented in an IR are valid for their specific plant. The applicant specific requirement to examine their operating history is intended to identify any technical basis which may have a time dependence for the evaluation of age-related degradation. If there are any plant specific time dependencies which would invalidate any of the IR conclusions, then the applicant would need to address those specific age-related degradation mechanism/ components combinations individually for the license renewal term. The IR requirement for an applicant to verify plant specific operating history is clarified in the Standardized Language.			
G-13	The use of linear extrapolation of cumulative fatigue usage factors to screen components may be unconservative and could exclude components which could suffer fatigue damage. The IR should evaluate and address this issue further.	The terminology "linear extrapolation of cumulative fatigue usage factors" must be defined to be extrapolation of number of transient cycles, if necessary, and the resummation of fatigue usage. Cumulative usage factors can only be extrapolated linearly when one set of transients dominates the fatigue evaluation. While this is often the case, the linear extrapolation of cumulative usage factors is, in general, unconservative. It should be noted that the number of assumed cycles that actually occur.	Closed	Linear extrapolation arguments have been removed from the IR.	It is believed that the response to this NRC comment fully resolves the concerns raised by the NRC. NUMARC requests that the NRC consider this comment "closed" based upon the response.

PWR INTERNALS IR: RESPONSE TO NRC COMMENTS

NRC COMMENT #	NRC COMMENT	INDUSTRY RESPONSE	STATUS	PROPOSED IR REVISION	REMARKS
G-14	The PWR internals listed in Table 3-3 (page 3-60) are evaluated as not safety significant because they do not perform any safety functions and are excluded from component life evaluation. See General Comment No. 4. While these components do not have an active safety function, they should be considered important to license renewal if the failure of these components as a result of age-related degradation has the potential of impacting the safety function of other safety significant components. Furthermore, the determination of component safety significance should also consider the impact of loose parts resulting from component failure on the safe operation of the reactor plant. There have been cases of partial failure of thermal shield supports - Maine Yankee and St. Lucie.	The PWR internals components listed in Table 3-3 (page 3-60)are considered to be out of the scope of the IR.	Closed	Revise report to clearly state that these components are not within the scope of the IR. Add a new Table 3.2 clearly identifying compoents within the scope of the report. Eliminate original Table 3-3. [p.3-8, Table 3.2]	It is believed that the response to this NRC comment fully resolves the concerns raised by the NRC. NUMARC requests that the NRC consider this comment "closed" based upon the response.
G-15	In Section 5 of component life evaluation, all safety significant internals components with no history of degradation were determined to be adequately managed in detecting and mitigating potentially significant age-related degradation. This determination in a large part relies upon the performance of normal inservice inspection (ISI) that cracks in the components resulting from age-related degradation will be detected. The ISI of these components based on the ASME Code, Section XI, requires only visual examination VT-3 on the accessible surfaces with a frequency of once every 10 years. The main objective of visual examination VT-3 does not appear to be adequate to find the cracks in these components.	The IR will be revised to identify clearly in Section 5 the program elements that manage the effects of potentially significant age-related degradation on PWR internals components. In some cases these program elements will be based wholly or in part on inservice examination requirements of the ASME Code Section XI, Subsection IWB. In other cases the program elements may be based on analytical evaluation of the potential for small flaws to grow to critical size under the loading and environmental conditions to which the component is subjected. The revisions will provide acknowledgement that the ASME Code bodies are preparing a set of inservice examination requirements specific to internals components through a new Section XI subsection (subsection IWG).	Closed	The IR will be revised in Section 5 to include a description of the current inservice examination requirements contained in ASME Section XI, Subsection IWB, along with a justification of the VT-3 examination procedures as they apply to accessible surfaces of internals components. A discussion of the proposed subsection IWG will be included in Section 5 of the IR. [p. 5-4, Section 5.1]	It is believed that the response to this NRC comment fully resolves the concerns raised by the NRC. NUMARC requests that the NRC consider this comment "closed" based upon the response.

PWR INTERNALS IR: RESPONSE TO NRC COMMENTS

NRC COMMENT #	NRC COMMENT	INDUSTRY RESPONSE	STATUS	PROPOSED IR REVISION	REMARKS
	To ensure adequate inspection of components susceptible to age-related degradation, VT-1 or surface examination of the threaded areas should be performed prior to end during the license renewal period, and at a frequency such that the potential crack growth will not exceed the allowed critical flaw sizes. Furthermore, the report did not address the structural integrity of internals components which are not accessible for normal ISI during refueling outages. The life evaluation of those inaccessible components should also be addressed.	This acknowledgement will in no way imply that current subsection IWB inservice examination requirements that are applied to internals components (examination category B-N-3)are inadequate. It is the intent of the visual examination requirements of the ASME Code that "crack-like" indications detected during VT-3 examinations represent a relevant condition. While there are some differences in the characteristics of VT-1 versus VT-3 examinations, such as demonstrated character recognition height and minimum distance from eye to object, both VT-1 and VT-3 as applied to the examination of accessible surfaces of internals components are capable of detection of surface flaws of significant length. When combined with analytical demonstration of tolerance to such flaws, the examination procedures are further justified. It should be noted that any reportable condition from VT-3 examination requires further characterization, using surface/volumetric examination or other means, in order to justify continued operation. Components that are inaccessible for inservice examination, and that cannot be rendered accessible by removal of the core or of other internals require plant-specific consideration, as described in Section 6 of the IR. Also, justifications for exemptions/reliefs from inservice examination requirements are a plant-specific consideration.		Plant-specific options for dealing with the examination of component surfaces that are not accessible, and which cannot be made accessible, will be provided in Section 6. [p. 6-4, Section 6.2.3]	

PWR INTERNALS IR: RESPONSE TO NRC COMMENTS

NRC COMMENT #	NRC COMMENT	INDUSTRY RESPONSE	STATUS	PROPOSED IR REVISION	REMARKS
G-16	It is well known that it takes time to initiate stress corrosion cracking and the cracking may be accelerated with the presence of sensitized materials, high tensile residual stresses resulting from cold working and welding, crevices, stagnant flow and the history of coolant contamination. The good operating experiences of austenitic stainless steel components in the primary water environment so far may not guarantee similar performance in the next 40 years. The IR should propose augmented inservice inspection of these components when the conditions that would promote SCC as mentioned above are present.	We agree that it takes time to initiate stress corrosion cracking (SCC) and that cracking may be accelerated by certain conditions. However, the IR clearly addressed the need for three conditions to promote SCC. The primary reason SCC is not deemed an age-related degradation mechanism for life extension is due to the primary coolant chemistry. In fact, much of the data generated by GE provides additional support as to the critical nature of oxygen in water affecting the corrosion potential, thus SCC initiation on internal components. It is expected that 40 years of operation or even 20 years does provide a high level of confidence regarding the PWR primary coolant, the metal lurgical conditions of the internals and the various states of stress and the potential for SCC. Augmented inspections are not warranted.	Open	The IR will be modified to include more discussion regarding the PWR environment. Specifically, BWR experiments which have demonstrated the importance of low oxygen and low halogens. will be added to section 4.0 of the IR to support that SCC is not a concern for license renewal. [p. 4-7, Para. 4]	It is believed that the response to this NRC comment fully resolves the concerns raised by the NRC. NUMARC requests that the NRC consider this comment "closed" based upon the response.

PWR INTERNALS IR: RESPONSE TO NRC COMMENTS

NRC COMMENT #	NRC COMMENT	INDUSTRY RESPONSE	STATUS	PROPOSED IR REVISION	REMARKS
G-17	Primary water stress corrosion cracking (PWSCC) of alloy 600 thermal sleeves were reported in several domestic PWR pressurizers. Therefore, component life evaluation for PWR internals should include components made of alloy 600.	No PWR internals components made of Alloy 600 have been identified in the report for any of the designs of the three domestic PWR vendors. Ni-Cr-Fe alloys are used for some components, notably for bolting and other mechanical fasteners. Stress corrosion cracking (SCC) of these Ni-Cr-Fe fasteners, as the result of a combination of material susceptibility, applied stress and reactor coolant environment, is described in some detail in Section 4.2 of the report. In particular, the 70Ni-16Cr-7Fe Alloy X-750 (SB-637, Grade N07750) is discussed on [illegible]. SCC (or PWSCC) was deemed t[illegible]otentially significant for fasteners m[illegible]e of this material, with further evaluation described in Section 5.3. While the discussion does not identify PWSCC, the issue is SCC of PWR internals. We believe that this covers the topic.	Closed	Should other Ni-base alloys be identified during the review and revision of the IR, the affected components will be described and the potential degradation evaluated in Sections 4.2 and 5.3. No revision t[illegible]e IR is needed to cove[illegible]iscussion of SCC (or PW[illegible] the Ni-Cr-Fe alloys id[illegible]o date. An extremely limited number of Alloy 600 subcomponents (1-2 subcomponents in 1-2 plant-[illegible]pecific designs) were identified [illegible]ring the review and revision of [illegible]he IR following the 4-15-92 NRC/NUMARC PWR Internals IR Meeting. This amount was insufficient to warrant addressing Alloy 600 generically (See note in Table 3.3) License renewal applicants will verify that no other Ni-Cr-Fe alloys, such as Alloy 600, are present in the reactor internals of their specific plant designs while applying the standardized "if, then" criteria in Section 4.2.3. No additional revision to the IR is needed to cover the discussions of PWSCC.	It is believed that the response to this NRC comment fully resolves the concerns raised by the NRC. NUMARC requests that the NRC consider this comment "closed" based upon the response.

PWR INTERNALS IR: RESPONSE TO NRC COMMENTS

NRC COMMENT #	NRC COMMENT	INDUSTRY RESPONSE	STATUS	PROPOSED IR REVISION	REMARKS
G-18	The IR often does not give the specifics on how certain conclusions were reached, or demonstrate that the processes proposed to manage degradation are actually feasible. For example, the primary tools for the management of the issues identified in Section 5 are inspection and fracture mechanics analysis, but neither the report nor the references seem to contain the scoping calculations that are necessary to demonstrate that the cracks of concern can in fact be detected with reasonable reliability by the inspection techniques available.	Because of the variation in internals configurations and loadings, it is impractical to include the specifics on the inservice examinations and fracture mechanics evaluations that have been or that could be used to demonstrate that potentially significant age-related degradation is adequately managed. However, it is the procedure, rather than the results of application of the procedure, that is at issue in Section 5 of the IR. This procedure consists of the following steps: (1) inservice visual examination at periodic intervals, with a requirement that all relevant conditions be reported and evaluated; (2) elastic or elastic-plastic fracture mechanics calculations to show that "crack-like" relevant conditions do not threaten the integrity of the component; and (3) plant-specific action for those cases that cannot	Closed	Section 5 of the IR will be revised to clearly delineate the steps in the programs used to manage the effects of potentially significant age-related degradation. For those cases where a combination of inservice examination and fracture mechanics can be used to manage degradation, the criteria for program element effectiveness will be described in Section 6.0. This will include citing reference flaw sizes, flaw growth curves, and margins against flaw instability contained in the ASME Code Section XI, relevant Code Cases, and relevant non-mandatory appendices. [p. 6-3, Section 6.2.2]	It is believed that the response to this NRC comment fully resolves the concerns raised by the NRC. NUMARC requests that the NRC consider this comment "closed" based upon the response.

NRC COMMENT #	NRC COMMENT	INDUSTRY RESPONSE	STATUS	PROPOSED IR REVISION	REMARKS
		be shown to conform to (1) and (2). The criteria for Step (1) are already contained in the ASME Code Section XI, including remedial actions such as repair and replacement. Criteria for Step (2) are not contained explicitly in the ASME Code Section XI, but can be adapted from related Code Cases and non-mandatory appendices of Section XI. Criteria for Step (3) must be justified by each applicant on a plant-specific basis. It is within the scope of the IR to describe the process in sufficient detail so that its effectiveness can be determined by the staff. It is beyond the scope of the IR to provide component-specific applications or results of application of the process steps.			
G-19	In general, the IR asserts that stress corrosion cracking of stainless steels is not a significant age-related degradation mechanism in PWRs (except for a few specific cases where cracking has occurred in service) because of the relatively benign environment. However, crevices are known to promote stress corrosion type cracking (even in the absence of high stress) in stainless steels and other austenitic alloys even when the bulk environment is of relatively high purity. The report should include discussions of components with crevices or creviced geometries and evaluate the potential for crevice corrosion cracking.	It is agreed that some discussion regarding creviced type geometries within the PWR internals is needed. Much of the available literature regarding the potential for crevices concentrating in a local environment is for oxygenated and/or oxidizing environments. There is no know data to suggest that PWR primary coolant type chemistries, e.g. low oxygen and low halogens, will promote SCC on creviced type geometries.	Open	section 4.9 of the IR will be modified to include a discussion on crevice corrosion. Unless data to the contrary is found, the potential for crevice corrosion degradation will be considered managed if proper primary coolant chemistry is maintained. [p. 4-24, Section 4.9.2]	It is believed that the response to this NRC comment fully resolves the concerns raised by the NRC. NUMARC requests that the NRC consider this comment "closed" based upon the response.

PWR INTERNALS IR: RESPONSE TO NRC COMMENTS

NRC COMMENT #	NRC COMMENT	INDUSTRY RESPONSE	STATUS	PROPOSED IR REVISION	REMARKS
G-20	The mechanical redundant feature is used in the IR to demonstrate acceptable performance of internal components having multiple supports during the license renewal term. However, it should be noted that age-related degradations are common mode failure mechanisms, and they could cause simultaneous failures in a multiple-support system. The IR lacks justification to support the assertion that PWR internals are not susceptible to common mode age-related failure mechanisms. The IR should be revised to address this issue. See General Comment No. 7.	The concern about common mode failure mechanisms is similar to Part (e) of Comment G-7. The variability within a bolted joint is considerable, with the examples of bolt hardness and bolt preload being typical. It is not unusual to see bolt hardness (a surrogate for yield strength) vary by 40% in a bolted joint, corresponding to a yield stress variation from approximately 100 to 130 ksi. Similarly, depending on the torquing procedure, bolt preload can vary by ±30% (four passes) to a factor of 2 (three passes). With such variability, only the application of an extreme (Level D) load would challenge a substantial number of bolts simultaneously. For nominal operating conditions, preferential degradation is expected.	Closed	See the proposed IR revision to comment G-7.	It is believed that the response to this NRC comment fully resolves the concerns raised by the NRC. NUMARC requests that the NRC consider this comment "closed" based upon the response.
G-21	Other than defining the interface boundary between internals and the pressure vessel, the report does not address age-related degradations at the pressure boundary regions. The IR should address potential problems at these regions.	The reactor vessel internals do not include the pressure boundary. The interfaces between the reactor internals and the pressure boundary (reactor vessel) are all readily inspectable, if needed.	Closed	Clarify IR scope in Section 2.0.	It is believed that the response to this NRC comment fully resolves the concerns raised by the NRC. NUMARC requests that the NRC consider this comment "closed" based upon the response.

PWR INTERNALS IR: RESPONSE TO NRC COMMENTS

NRC COMMENT #	NRC COMMENT	INDUSTRY RESPONSE	STATUS	PROPOSED IR REVISION	REMARKS
G-22	Materials for major PWR internals are specified as austenitic stainless steel or nickel-based alloys. This includes a large group of materials and their susceptibility to age-related degradations may vary. Effects of aging degradations on core internals cannot be evaluated without knowing the specific materials of construction. The IR should be revised to clearly identify the material and the potential age-related degradation mechanisms for each major PWR internals.	It is understood that the specific materials of construction, for each component for the three vendors various designs would be too cumbersome for this IR. It is also noted that in considering material degradation, material categories should be sufficient. A specific category will cover materials of similar chemical composition. More detail, in the form of material categories for the internals will be added.	closed	A table will be added to Section 3.0 of the IR to address material categories used in the construction of PWR internals. Each category will be briefly discussed in the text of the IR and examples of each category will be provided, e.g. Austentic Stainless Steels would be a category for materials such as Types 304 and 316. [p. 3-19, Table 3.3]	It is believed that the response to this NRC comment fully resolves the concerns raised by the NRC. NUMARC requests that the NRC consider this comment "closed" based upon the response. This comment is similar to comment G-11.
S-1	Both in-reactor tests and field experience with stainless steel fuel cladding have shown that irradiation-assisted-stress-corrosion cracking (IASSC) can occur in PWRs. This may indicate either that the local environments can be sufficiently aggressive to produce SCC problems or that the cracking does not require an aggressive, aqueous environment. Impurity segregation grain boundaries in some low-purity materials (in particular Type 304 SS) may be so extensive at high fluence that the materials will crack even in inert environments if the stress level is sufficiently high. Until the mechanism of IASCC is better understood and it can be shown that the electro-chemical potential is low throughout the core and upper internal region, all stainless steels with fast fluence $5x10^{20}$ n/m^2 should be considered potentially susceptible to IASCC.	We agree that irradiation-assisted stress corrosion cracking (IASCC) occurs in stainless steel cladding of PWR fuel elements. Detailed analysis of these situations has shown that local stress states caused by fuel relocation or pellet-clad interaction were the significant factor. An adequately severe combination of susceptible material (i.e., irradiated beyond the threshold of $5x10^{20}$n/cm^2), high stresses and an aggressive environment is needed. For components irradiated to levels below $5x10^{20}$n/cm^2 and exposed to PWR coolant conditions, the stresses must be quite large or the component must contain flaws of some size. BWR internals components have been evaluated using degraded material properties caused by exposure to fluences of $5x10^{20}$n/cm^2 (ductility and elastic-plastic fracture toughness) and with design stresses. Postulated flaws do not lead to applied stress intensities sufficient to cause crack initiation. Based on these screening calculations, the criterion that:	Closed	Section 4.3 will be revised to clarify the threshold levels and criteria used to support the finding of non-significance. [p. 4-11, Section 4.3.2]	It is believed that the response to this NRC comment fully resolves the concerns raised by the NRC. NUMARC requests that the NRC consider this comment "closed" based upon the response.

NRC COMMENT #	NRC COMMENT	INDUSTRY RESPONSE	STATUS	PROPOSED IR REVISION	REMARKS
		1. exposure is less than $5x10^{20}n/cm^2$; 2 the environment is controlled to PWR operating limits; and 3. the operating stresses are within the design envelope. Leads to a finding of non-significance. For stress levels that are moderate to high, the threshold for significant exposure is $1x10^{20}n/cm^2$.			
S-2	Page 1-21, top paragraph: The IR states that another set of component-degradation mechanism combinations is considered to be generically resolved because the potentially significant age-related degradations can be shown to be bounded within established limits by currently acceptable maintenance, inspection, test, and analytical evaluation procedures which are in place at the various utilities. How could the potentially significant mechanisms be resolved generically without plant-specific assessment of the degradation levels and management programs? The IR should either provide a detailed generic discussion or keep these issues open for resolution on a plant specific basis.	The industry reports separate generic conclusions from any plant-specific actions by using a standardized format (e.g., Section 4 evaluates the potential significance of age-related degradation mechanism), standardized language that outlines license renewal applicant responsibilities, and definitive criteria for drawing generic conclusions. The intent of Section 4 for example, is to provide criteria that, if met by the applicant, represent a generic basis for the determination that a particular age-related degradation mechanism is insignificant for a particular component. The intent of Section 5 is to provide an adequate description of program elements that, if in effective use by the applicant, represent a generic basis for management of potentially significant degradation. Failure to satisfy the criteria in Section 4, or to have the program elements of Section 5 in place, requires the applicant to perform a plant-specific evaluation.	Closed	The IR will be modified to incorporate the standardized language providing an outline of license renewal applicant responsibilities relative to report findings.	It is believed that the response to this NRC comment fully resolves the concerns raised by the NRC. NUMARC requests that the NRC consider this comment "closed" based upon the response.

PWR INTERNALS IR: RESPONSE TO NRC COMMENTS

NRC COMMENT #	NRC COMMENT	INDUSTRY RESPONSE	STATUS	PROPOSED IR REVISION	REMARKS
S-3	Page 1-25, top paragraph: Regarding the support columns, the IR states that the components are subject to limited crack growth, because of the presence of compressive stress apparently during normal operation. It is not clear whether an assessment has been made for possible existence of tensile stresses during other plant conditions. This should be verified and confirmed in the IR.	It appears the NRC concern is regarding the variation in the loading conditions of the support columns. While it is generally true that these components experience compressive stresses over most of their life, there may be short periods of time where some components, on a plant-specific basis, will experience tensile stresses due to thermal transients or design basis events. A detailed study of this issue on a generic basis is outside the scope of the IR. However, crack growth of an existing flaw would be limited due to the short duration of the accident stresses and hence is inconsequential. The discussion on cracking and its growth will be removed from the top of page I-25.	Closed	See proposed IR revision to comment S-40.	It is believed that the response to this NRC comment fully resolves the concerns raised by the NRC. NUMARC requests that the NRC consider this comment "closed" based upon the response.
S-4	Page 1-26, second and third paragraph: Regarding fatigue evaluation, the IR states that re-analysis and an acceptable in-service inspection program would be carried out, and as a result it concludes that fatigue is not a significant age-related degradation mechanism. A degradation mechanism cannot be eliminated generically at present based on plant-specific results of future re-analysis and inspection programs. The IR should be based on results which are known at present. In addition, if the results are plant-specific, the IR should describe plant-specific evaluation and should not dismiss the degradation mechanism generically. Fatigue is an open issue.	Section 5 of the IR describes the elements of a fatigue analysis/reanalysis and inservice inspection program for fatigue-critical internals that, if shown by a license renewal applicant to be in effective use, provides a generic basis for management of potentially significant fatigue damage. The IR is not required to report the results of an established program application; however, the IR does report the results of the application of the B&W fatigue reanalysis (see IR References 4-3, 4-37, and 4-38) in order to demonstrate the effective nature of the process. Such a demonstration gives a measure of confidence to license renewal applicants that the use of fatigue reanalysis will result in low fatigue usage factors for the license renewal term.	Closed	None	In agreement between NUMARC and the NRC staff at the 4-15-92 NRC/NUMARC PWR Internals IR meeting, Comment S-4 is "closed" for this IR. Fatigue remains an "open" issue as noted in comment S-27.

PWR INTERNALS IR: RESPONSE TO NRC COMMENTS

NRC COMMENT #	NRC COMMENT	INDUSTRY RESPONSE	STATUS	PROPOSED IR REVISION	REMARKS
S-5	Page 3-5 and 3-45: The IR lists a few reactor components that are not included in the subject IR. This IR should provide references to other pertinent IRs that address the excluded components. This will ensure that the excluded components have not been missed.	Components that are not included are out of the scope of the IR: cross-references to other IRs will not be provided	Closed	Assure clarity of scope in Section 2.0	It is believed that the response to this NRC comment fully resolves the concerns raised by the NRC. NUMARC requests that the NRC consider this comment "closed" based upon the response. Related to Comment G-4.
S-6	Page 3-27, fourth paragraph: The IR generically eliminated the need for evaluation of the Westinghouse secondary core support assembly (and similarly the Combustion Engineering core stop interface) based on the assertion that its function is to provide core support in the unlikely event of core barrel failure. It is not justifiable to accept failure of such an important component, though redundant, and lose the existing safety margin provided with the original design. The degradation of this component should be evaluated for license renewal. See General Comment No. 7.	The IR will be revised to indicate that the secondary core support is a component which is out-of-scope of the IR.	Closed	Section 2 of the IR will be revised to indicate that the secondary core support is a component which is out-of-scope of the IR.	It is believed that the response to this NRC comment fully resolves the concerns raised by the NRC. NUMARC requests that the NRC consider this comment "closed" based upon the response.

PWR INTERNALS IR: RESPONSE TO NRC COMMENTS

NRC COMMENT #	NRC COMMENT	INDUSTRY RESPONSE	STATUS	PROPOSED IR REVISION	REMARKS
S-7	Page 3-39, last paragraph: Regarding the hold-down spring the IR concludes that the component is not safety-significant, and no further evaluation is required. However, the hold-down spring is needed to provide vertical core restraint. Moreover, degradation of the hold-down spring has been observed in past inspections resulting in wearing of the reactor vessel flange and head (Page 3-54). Therefore, it does not seem appropriate to classify the hold-down spring as non-safety-significant and not to evaluate the degradation effects. The IR should address this issue.	The IR used the term "not safety significant" to imply that component failure would not adversely affect the ability to shut down the plant or to maintain a coolable geometry. The holddown springs meets this test. The only postulated degradation mechanisms for the holddown spring is relaxation. There currently exists methods and standards for monitoring of internals for holddown spring relaxation. In addition, visual inspections and preload measurements can be readily performed, if a need is determined to exist at a particular plant. The rationale for eliminating holddown springs as safety insignificant is provided on page 3-39, where it is also noted that other components are available to take any loads a degraded holddown spring does not. However, the IR will be revised to not make a determination on whether holddown springs are safety significant or important to license renewal. Holddown springs will be outside the scope of the IR.	Closed	The IR will be revised so that holddown springs are outside the scope of the IR.	It is believed that the response to this NRC comment fully resolves the concerns raised by the NRC. NUMARC requests that the NRC consider this comment "closed" based upon the response.
S-8	Page 3-40 to 3-44, Sections 3.1.2.4 to 3.1.2.7: Thermal shields, incore instrumentation guide tubes, irradiation specimens guides and inlet flow distribution components are all included as elements in the three generic regions for core internals as given in Tables 3-2 (a-c). In order to have a consistent report format and to avoid unnecessary confusions, these components should be included as subsections to the three generic core regions.	We agree with comment. However, the report will no longer assess whether these components are important to safety. They will either be in or out of the scope of the IR. Note that thermal shields, the non-pressure boundary portions of the instrument guide tubes, the irradiation specimens guides, and inlet flow distribution components will be considered outside the scope of the IR.	Closed	Modify section 2 of the IR to remove thermal shields, the instrument guide tubes, the irradiation specimen guides, and the inlet flow distribution components from the scope of the IR.	It is believed that the response to this NRC comment fully resolves the concerns raised by the NRC. NUMARC requests that the NRC consider this comment "closed" based upon the response.

PWR INTERNALS IR: RESPONSE TO NRC COMMENTS

NRC COMMENT #	NRC COMMENT	INDUSTRY RESPONSE	STATUS	PROPOSED IR REVISION	REMARKS
S-9	If the age-related degradation could result in the loss of the surveillance samples, the guides must be considered for component life evaluation. Page 3-43, Section 3.1.2.6 Irradiation Specimen Guides: The IR classifies the irradiation specimen guides and holders as non-safety-significant. Failure of these components may result in mismanagement of the irradiation specimens which are required by regulations and are fundamental to monitoring the fluence level of the reactor pressure vessel (RPV) and other components. A reliable irradiation estimate of the RPV is needed for its life evaluation. The IR should evaluate and address these components.	The IR will be revised to indicate that irradiation specimen guides are out-of-scope of the IR.	Closed	The IR will be revised to indicate that irradiation specimen guides are out-of-scope of the IR.	It is believed that the response to this NRC comment fully resolves the concerns raised by the NRC. NUMARC requests that the NRC consider this comment "closed" based upon the response.
S-10	Page 3-45, first paragraph, last sentence: The IR states that although the majority of PWR internals were fabricated prior to implementation of Subsection NG, the design philosophies of the three PWR vendors were such that the intent of Subsection NG was met. Certain requirements of Subsection NG might not have been met. The IR should include a discussion identifying the NG requirements which were not satisfied by the earlier designs and evaluate its impact on license renewal.	The reviewers' concern about potentially not having met all NG requirements applies equally to the current licensing basis as well as to the license renewal period. Since the current licensing basis has been deemed adequate, it is not obvious that further effort is required to address this comment. However, since no credit was taken for this statement, the IR will be revised to delete this statement in order to minimize future confusion.	Closed	Delete the indicated statement. [p. 3-23, Para. 2]	It is believed that the response to this NRC comment fully resolves the concerns raised by the NRC. NUMARC requests that the NRC consider this comment "closed" based upon the response.

PWR INTERNALS IR: RESPONSE TO NRC COMMENTS

NRC COMMENT #	NRC COMMENT	INDUSTRY RESPONSE	STATUS	PROPOSED IR REVISION	REMARKS
S-11	Pages 3-46 and 3-47, items a,b,c: Regarding the earthquake forces, the IR uses the terms design and hypothetical apparently to indicate operating and safe shutdown, respectively. If so, the IR should be revised by using the currently acceptable terminology.	We agree with the comment and will revise IR to use standard terminology (OBE and SSE).	Closed	Revise IR to use standard terminology (OBE and SSE). [p. 3-24, Para. 3]	It is believed that the response to this NRC comment fully resolves the concerns raised by the NRC. NUMARC requests that the NRC consider this comment "closed" based upon the response.
S-12	Section 3.2.3.1 Leak-Before-Break: For plants approved for leak-before-break with piping fabricated from cast stainless steel, the effect of thermal aging on cast stainless steel should be evaluated to ensure that the leak-before-break acceptance criteria would be satisfied throughout the license renewal term.	We believe that this comment is relevant to the reactor coolant systems IR and not the internals. Leak-Before-Break (LBB) is presented as an example of analysis which, if performed on a plant's piping, can result in reduced loads on the internal components. It is agreed that if a plant has LBB for its current license, then that evaluation may need to be resubmitted for the license extension period.	Closed	None	It is believed that the response to this NRC comment fully resolves the concerns raised by the NRC. NUMARC requests that the NRC consider this comment "closed" based upon the response.
S-13	Section 4.1 Irradiation Embrittlement: This section indicates that irradiation embrittlement reduces that ductility properties, but it is not considered as a significant age-related degradation mechanism. The IR has not provided a fracture mechanics analysis to demonstrate that the PWR internals have adequate toughness after they have had their ductility reduced. The IR should provide such a demonstration by using the fracture toughness properties for weld metal irradiated to the end of license renewal period. A critical flaw size should be determined and IR should recommend an inspection method for detection of flaws and an inspection period to ensure flaws do not grow to a critical size. See General Comment No. 15.	A generic fracture mechanics analysis is outside the scope of the IR. However, the use of fracture mechanics, coupled with inspection, is a viable aging management tool on a plant-specific basis.	Closed	Remove discussion from Section 4.0 and change Section 6 to address fracture mechanics and inspections on a plant specific basis. [p. 6-3, Section 6.2.2]	It is believed that the response to this NRC comment fully resolves the concerns raised by the NRC. NUMARC requests that the NRC consider this comment "closed" based upon the response. Also, see responses to comments S-14e, G-9, and G-15.

PWR INTERNALS IR, RESPONSE TO NRC COMMENTS

NRC COMMENT #	NRC COMMENT	INDUSTRY RESPONSE	STATUS	PROPOSED IR REVISION	REMARKS
S-14	Page 4-4 through 4-7: (a) The IR presents experimental data to show how material properties change after prolonged exposure to neutrons. The data indicate that for the irradiated material, the yield strength approaches the ultimate strength and the uniform elongation and the total elongation drop substantially from their unirradiated values. However, the IR concludes that there still remains a significant amount of ductility. It appears that the data do not support the conclusions. In addition, the most important parameter of concern for evaluation of cracked components is the fracture toughness which is substantially reduced due to neutron embrittlement. The IR should provide more details about the experiments (e.g., fluence, flux and temperature for BWR; fracture toughness and tearing modulus for PWR), and reevaluate by considering the appropriate materials and the end-of-life fluence level at a realistic flux level.	This IR, and a related IR covering BWR internals components, present experimental data on mechanical properties of irradiated austenitic stainless steel, showing that the yield strength increase and the elongation decreases substantially. However, the issue is not the magnitude of change, but whether the residual mechanical properties are adequate to assure structural integrity of the component during the license renewal term. Fracture toughness data for irradiated stainless steel also indicates substantial reduction, from very high elastic-plastic J_{IC} values of 2000 to 5000 in-lb/in^2 and very steep J-R curves. With irradiation to high fluences (>10^{21}n/cm^2, E > 0.1 Mev), J_{IC} is reduced to values on the order of 100 to 200 in-lb/in^2. The tearing resistance is also degraded. These data have been obtained under conditions representative of BWR and PWR service conditions, and indicate a saturation effect at the higher fluence levels that mitigates concerns about end-of-life fluence. More data will be accumulated on mechanical properties of irradiated stainless steels in the future that will improve our understanding of accelerated aging effects. However, the present data base is adequate for the purpose of determining those components that are subject to potentially significant neutron irradiation embrittlement.	Closed	(a) Section 4.1.1 will be revised to include a discussion of mechanical property changes as a result of neutron exposure, including changes in fracture toughness and ductility. The section will also be revised to include a qualitative discussion of weld metal chemistry, and its effect on embrittlement, and of the potential for damage to non-stainless steel bolting, due to its location within the reactor vessel. [p. 4-3, Section 4.1.1]	It is believed that the response to this NRC comment fully rresolves the concerns rrasied by the NRC. NUMARC requests that the NRC consider this comment "closed" based upon the response.

PWR INTERNALS IR: RESPONSE TO NRC COMMENTS

NRC COMMENT #	NRC COMMENT	INDUSTRY RESPONSE	STATUS	PROPOSED IR REVISION	REMARKS
	(b) The materials in the heat affected zone (HAZ) usually suffer severe neutron embrittlement. The IR did not address this issue. The IR should consider the irradiation embrittlement effect on the HAZ materials.	The materials that comprise PWR internals are, with the exception of some bolting, austenitic stainless steel and Ni-Fe-Cr (nickel-base) alloys. Weld metals for these materials do not contain the trace constituents that would enhance irradiation embrittlement over and above that observed for base metal.	Closed	(b) None	
	(c) Certain parts of the internal components (e.g., bolts, welds, other attachments) are probably not made of stainless steel. The IR does not address neutron embrittlement of Ni-Cr-Fe components and should do so.	Only stainless steel or Ni-Cr-Fe bolting is used in the high flux regions of the core. Stainless steel bolting is discussed in Section 4.1 of the report. Ni-Cr-Fe bolting is not discussed in the report and a discussion will be added.	Closed	(c) Revise Section 4.1.2 as follows: "Irradiation embrittlement can occur in PWR internals compoents, such as these manfactured from stainless steel and Ni-Cr-Fe alloys, with expected neutron fluences in excess of 1 $*10^{20} n/cm^2$ (E>1MeV)." [p. 4-4, Para. 3].	
	(d) For fracture mechanics analyses, the IR refers to Sections 5.1.1.3 and 5.1.4.1.1. However, the IR does not contain these sections. Section 5.1.1.1.2 which deals with analysis of flaws should contain quantitative description of the influence of PWR internal environment on fracture toughness and crack growth rates for life extension purposes.	The reviewer has correctly identified typographical errors. As noted in the response to S-14(a), the IR will be revised to include more extensive discussion of irradiation-induced mechanical property changes in Section 4. The use of these data for aging management evaluations is covered under the response to S-14(e).	Closed	(d) The IR will be revised to cover fracture mechanics analyses in Section 6.2.2. [p. 6-3, Section 6.2.2]	

NRC COMMENT #	NRC COMMENT	INDUSTRY RESPONSE	STATUS	PROPOSED IR REVISION	REMARKS
	(e) The IR states that irradiation embrittlement of PWR internals materials in itself is not a significant aging degradation mechanism, since fracture mechanics analyses can be used for assessment. Availability of an assessment method alone is not an adequate ground for elimination of a degradation mechanism. The IR should provide a fracture mechanics analysis (either generic, i.e., enveloping all plant conditions or specifying that licensees referencing the IR in a renewal application must perform a plant-specific analysis) and a subsequent evaluation based on the analysis results.	For those internals components subject to potentially significant neutron irradiation embrittlement, Section 5 of the IR provides a description of program elements aimed at managing this form of degradation. These program elements consist of three steps: (1) inservice visual examination, in accordance with the requirements of the ASME Code Section XI, Subsection IWB; (2) more extensive volumetric or surface examination, in accordance with Code provisions, in order to characterize reported relevant conditions and their disposition; and (3) repair and/or replacement of those components subject to damage beyond that for which continued operation can be justified. The evaluation of "crack-like" relevant conditions consists of flaw growth calculations and a determination of continued structural integrity on a plant-specific bases. It is beyond the scope of the IR to provide generic or bounding results for such flaw evaluations for the wide variety of internal component designs, loadings and operating conditions. It is appropriate in the IR, however, to describe the evaluation process. The periodic visual examination part of the process (VT-3) is described sufficiently in the Code to require no further discussion here. Supplementary visual, surfaces and volumetric examination procedures to characterize relevant conditions (e.g., a crack-like indication) are also described in the Code, with the utility required to justify the particular procedures employed to jurisdictional authorities. It would be inappropriate for the IR to prioritize particular	Closed	None	

NRC COMMENT #	NRC COMMENT	INDUSTRY RESPONSE	STATUS	PROPOSED IR REVISION	REMARKS
		supplementary examination steps. In lieu of supplementary examination, a utility might choose to conservatively characterize the relevant condition and demonstrate directly, without further examination, that the safety function of the component is not compromised. Such a plant-specific action is permissible in the Code and should not be precluded by the IR. Similarly, in lieu of supplementary examination, a utility might choose to repair or replace the affected component. Again, such a plant-specific action is permissible in the Code and should not be precluded by the IR. The IR does provide guidance with respects to evaluation process steps, and these steps will be stated clearly in Section 5. For example, while the justification for fatigue crack growth rates and fracture toughness of irradiated material is the responsibility of the license renewal applicant, it is expected that irradiation will have little effect on fatigue crack growth rates in austenitic stainless steels.			

NRC COMMENT #	NRC COMMENT	INDUSTRY RESPONSE	STATUS	PROPOSED IR REVISION	REMARKS
		Therefore, relatively modest adjustments to the fatigue crack growth rate curve in Appendix C of Section XI will be required. More important than the effect of irradiation is the effect of the water environment, which can be accommodated by introducing an environmental factor consistent with in-progress work in support of Appendix C. The effect of irradiation on fracture toughness is substantial, but the importance is mitigated by low stress levels, compressive loadings, and small flaw sizes. Each plant-specific application of the flaw evaluation process will have to provide justification of the fracture toughness used.			
	(f) The IR concludes that in the absence of flaws, irradiation embrittlement is not a significant age-related degradation mechanism for the license renewal period, and no further evaluation is required. Since a flaw can be initiated due to other degradation mechanisms, the synergistic effect should be considered and the irradiation effect should not be dismissed from consideration for license renewal.	(f) Synergistic effects are included in the evaluation process, to the extent that service experience and laboratory data can justify. The visual examination is directed toward relevant conditions from any of a number of age-related degradation mechanisms, as are supplementary examinations The flaw evaluation is directed to... a combination of cyclic crack growth, possibly accelerated by environmental effects, and flaw stability, which can be affected by other age-related degradation processes. We believe that synergistic effects have been included in the evaluation process, to the extent necessary.Irradiation effects are not dismissed from further consideration in the IR. Section 4 prescribes criteria that must be met in order that neutron irradiation embrittlement of internals components is not significant. For those components that fail to meet these criteria, neutron irradiation embrittlement is potentially significant and the aging management programs in	Closed	(f) None	

PWR INTERNALS IR: RESPONSE TO NRC COMMENTS

NRC COMMENT #	NRC COMMENT	INDUSTRY RESPONSE	STATUS	PROPOSED IR REVISION	REMARKS
		Section 5 of the IR can be used for evaluation. For those license renewal applicants unable to demonstrate that the aging management programs in Section 5 are in place, plant-specific action are necessary. Some options in this regard are described in Section 6.			

PWR INTERNALS IR: RESPONSE TO NRC COMMENTS

NRC COMMENT #	NRC COMMENT	INDUSTRY RESPONSE	STATUS	PROPOSED IR REVISION	REMARKS
S-15	Page 4-6, Section 4.2.1: The IR correctly points out that the oxygen level during operation and the halogen levels are low for occurrence of stress corrosion cracking (SCC). However, the sulfate from demineralizer resins and the oxygen level prior to and during shutdown are potential contributors to SCC. The IR should evaluate these two sources as well.	The initial part of the comment regarding oxygen levels prior to start-up is correct and the IR will include a discussion on this. The second part of the comment regarding sulfate excursions from the demineralizers is not a common plant condition. This can occur as a result of a plant specific situation. It is noted that changes in primary coolant pH and conductivity, beyond a plants technical specification limits will prompt evaluations to establish the source of the changes and corrections will be made.	Closed	Section 4.2 of the IR will be modified to include a discussion on the oxygenated condition s which exist prior to start-up. The IR will reflect that halogen concentration must be maintained below 150 ppb at all times regardless of system temperature. Although oxygenated conditions may exist prior to start-up, halogens are kept below 150 ppb and the temperature can not exceed 180^0F. A cold, oxygenated water with halogens below 150 ppb is not an environment which will promote SCC on PWR internal component materials. [p. 4-7, Para. 3]	It is believed that the response to this NRC comment fully resolves the concerns raised by the NRC. NUMARC requests that the NRC consider this comment "closed" based upon the response.
S-16	Pages 4-7 to 4-9, Section 4.2.2: It is stated that stagnant conditions inside the pressure vessel could create a local environment which differs from the bulk PWR environment. There is no assurance that components made of type 304 stainless steel are not exposed to such a locally corrosive environment. The IR should be revised to address this potential degradation environment.	The sentence the reviewer is referring to is not applicable to internals and will be deleted from the IR. Crevice type corrosion will be discussed. Much of the available literature regarding the potential for crevices concentrating in a local environment, causing SCC on stainless steels, is for oxygen and/or oxidizing environments. There is not any known data to suggest that PWR primary coolant type chemistries, e.g. low oxygen and low halogens, will promote SCC on creviced type geometries within the PWR internals.	Open	The sentence the reviewer is referring to will be deleted. [p.4-6, Para. 2] A discussion of crevice corrosion will be added to section 4.9. of the IR. [p. 4-23, Section 4.9.2]	It is believed that the response to this NRC comment fully resolves the concerns raised by the NRC. NUMARC requests that the NRC consider this comment "closed" based upon the response. See also the response to comment G-19.

PWR INTERNALS IR: RESPONSE TO NRC COMMENTS

NRC COMMENT #	NRC COMMENT	INDUSTRY RESPONSE	STATUS	PROPOSED IR REVISION	REMARKS
S-17	Page 4-9: Radiation hardening can make a material more susceptible to stress corrosion cracking. The IR should address this potential degradation environment.	Reviewer is referring to IASCC degradation which is addressed in Section 4.3 of the IR. We believe this is adequately addressed in the IR.	Closed	None	It is believed that the response to this NRC comment fully resolves the concerns raised by the NRC. NUMARC requests that the NRC consider this comment "closed" based upon the response.
S-18	Page 4-10: Components fabricated from alloy 600 (e.g., flux thimble tubes) are potentially susceptible to stress corrosion cracking in the PWR coolant environment. The IR should address this potential degradation mode.	Westinghouse flux thimble tubes are made of austenitic stainless steel, not Ni-Cr-Fe materials such as Alloy 600. Combustion Engineering and Babcock & Wilcox in-core instrumentation guide tubes are also made of austenitic stainless steel. No Alloy 600 PWR internals components were identified during the preparation of this report. This finding will be reviewed in light of the staff comment. If such components are identified, Section 4 of the IR will be modified to include a discussion of PWSCC and its potential significance to PWR internals.	Closed	An extremely limited number of Alloy 600 subcomponents (1-2 subcomponets in 1-2 plant-specific designs) were identified during the review of the IR following the 4-15-92 NRC/NUMARC PWR internals IR meeting. This amount was insufficient to warrant addressing Alloy 600 generically (see note in Table 3.3). License renewal applicants will verify that no additional Ni-based alloys, beyond those currently discussed in Section 4.2, are present in the reactor internals of their specific plant designs while applying the standard "if, then" criteria in Section 4.2.3.	It is believed that the response to this NRC comment fully resolves the concerns raised by the NRC. NUMARC requests that the NRC consider this comment "closed" based upon the response See Comment G-17 for additional information.

PWR INTERNALS IR: RESPONSE TO NRC COMMENTS

NRC COMMENT #	NRC COMMENT	INDUSTRY RESPONSE	STATUS	PROPOSED IR REVISION	REMARKS
S-19	Page 4-11 and 4-12: Stress corrosion cracking of a large number of components has been eliminated from further consideration for license renewal apparently based on operating experience with these components for the last 20-30 years. In order to make a case for license renewal, the IR should demonstrate that these components will not be affected by SCC through the license renewal period.	Because there has been 20-30 years of PWR internals operating experience without the effects of SCC we expect similar behavior during the license renewal period. Laboratory tests to simulate the PWR environment for durations of the same order of magnitude are unnecessary and impractical. Operating experience coupled with maintenance of the primary coolant chemistry within the limits specified throughout the current licensing period will provide adequate SCC-management of internals components. There is not any known data which support a contrary position and , in fact, much of the work performed by GE to justify (HWC) hydrogen water chemistry has also demonstrated the importance of low oxygen, low halogen water.	Closed	Section 4.2 of the IR will be modified to more explicitly define the primary coolant chemistry limits and the significance of maintaining these limits throughout the license renewal period. [P. 4-7, Section 4.2.2]	It is believed that the response to this NRC comment fully resolves the concerns raised by tne NRC. NUMARC requests that the NRC consider this comment "closed" based upon the response. Also, see comments G-I6, G-19, S-15 and S-16.
S-20	Section 4.3: Irradiation-Assisted Stress Corrosion Cracking (IASCC): This section indicates that IASCC is not an age-related degradation mechanism for the license renewal period because of the relatively benign environment of PWR coolant, which incorporates hydrogen overpressure. This conclusion needs to be justified.	Irradiation-assisted stress corrosion cracking (IASCC) is determined to be insignificant for PWR internals on the basis of a favorable combination of these factors: 1. the relatively benign PWR coolant environment, including the hydrogen overpressure; 2. The relatively low stress levels seen by internals during normal operation;	Closed	Section 4.3 the IR will be revised to clarify of the criteria that must be satisfied in order for a license renewal applicant to take credit for the generic conclusion. [p. 4-7, Section 4.3.2]	It is believed that the response to this NRC comment fully resolves the concerns raised by the NRC. NUMARC requests that the NRC consider this comment "closed" based upon the response. Related to Comment S-1.

NRC COMMENT #	NRC COMMENT	INDUSTRY RESPONSE	STATUS	PROPOSED IR REVISION	REMARKS
		3. the residual material ductility and fracture toughness following radiation exposure. For low stresses (<10 ksi), fluence up to $5x10^{20}n/cm^2$ is acceptable in a controlled PWR coolant environment. For higher stresses (>10 ksi), the fluence should be limited to $1x10^{20}n/cm^2$. If these thresholds are exceeded, a plant-specific fracture mechanics and flaw tolerance evaluation is needed to determine a surface, visual or volumetric inspection requirement.			
S-21	Page 4-14, Section 4.4.2: The IR states that the reactor coolant fluid velocities are well below those which are required for erosion. Since the fluid velocity is a suspected cause for the observed thinning of thimble tubes, the IR should address this issue.	Fluid velocities do not cause erosion in flux thimble tubes but can, in a limited number of instances, cause wear through the action of flow-induced vibration. Thimble tube wear is discussed in Section 4.7 and 5.5 of the IR.	Closed	None. Thimble tubes are outside the scope of the IR.	It is believed that the response to this NRC comment fully resolves the concerns raised by the NRC. NUMARC requests that the NRC consider this comment "closed" based upon the response. See also, the response to comment G-5
S-22	Page 4-15, Section 4.5.2: The IR states that for the austenitic stainless steels and Ni-base alloys of which most PWR internals components are constructed, creep is not a concern. However, the IR did not identify materials for the remaining components nor justify why creep is not significant for these components. This information should be provided in the IR.	The reactor internals components are constructed from austenitic stainless steels, Ni-base alloys, precipitation-hardened austenitic stainless steels, cobalt base alloys and martensitic stainless steels. There are no carbon steel PWR reactor vessel internal components. For the materials of construction, creep is not a concern as the operating temperatures are below 1000^0F.	Closed	A list of categories of materials used in the construction of the PWR internal components will be added to Section 3.0 of the IR. Section 4.5.2. will also be modified to include the entire list of material categories. [p. 3-9, Table 3.3, p. 4-15, Para 2]	It is believed that the response to this NRC comment fully resolves the concerns raised by the NRC. NUMARC requests that the NRC consider this comment "closed" based upon the response. See also, the response to comment G-10.

PWR INTERNALS IR: RESPONSE TO NRC COMMENTS

NRC COMMENT #	NRC COMMENT	INDUSTRY RESPONSE	STATUS	PROPOSED IR REVISION	REMARKS
S-23	The IR did not address the issue of managing relaxation damage in inaccessible regions where UT detection is not possible. In addition, the reliability of the UT detection is questionable. Additional information concerning inaccessible regions and reliability of the UT detection should be provided in the IR.	Section 4.6 and 5.4 do not rely on UT techniques to address aging management of stress relaxation; consequently, there is no need to discuss reliability of UT techniques. As discussed in section 6.1.1. However, UT inspection techniques are considered to be a potentially viable method for providing aging management. Aging management programs which employ UT inspections would necessarily have to demonstrate their reliability. As discussed in the response to NRC Comment G-9, a list of components which cannot be inspected by the current and effective procedures in Section 5 can be made. Any component which could not be inspected by these methods would be referred to Section 6 for aging management.	Closed	Proposed changes in response to comment G-9 serve to address this comment.	It is believed that the response to this NRC comment fully resolves the concerns raised by the NRC. NUMARC requests that the NRC consider this comment "closed" based upon the response. See also, the response to comment G-9.

NRC COMMENT #	NRC COMMENT	INDUSTRY RESPONSE	STATUS	PROPOSED IR REVISION	REMARKS
S-24	Section 4.6 - Stress Relaxation		Closed	None	It is believed that the response to this NRC comment fully resolves the concerns raised by the NRC. NUMARC requests that the NRC consider this comment "closed" based upon the response. See also, the response to comments G-9, G-10 and S-23.
	(a) This section cites eight references (Reference 4-21 through 4-28) in its discussion on the significance of stress relaxation to reactor internal components. Provide copies of those references, which contain stress relaxation information relative to Babcock and Wilcox and Westinghouse Baffle-Former Assembly Bolts and Combustion Engineering Core Shroud Assembly Bolts.	(a) These requested references will be provided under a separate cover.			
	(b) To assist the staff in its review of this issue, provide more detailed drawings or sketches of the Babcock and Wilcox and Westinghouse Baffle-Former assembly and Combustion Engineering Core Shroud Assembly.	(b) The IR recognizes stress relaxation as a Section 6 aging management issue for Babcock and Wilcox and Westinghouse baffle former assembly and Combustion Engineering core shroud assembly. As discussed in Comment G-9 and S-23, stress relaxation is a Section 6 aging management concern only for components which cannot be inspected by the current and effective inspection methods discussed in Section 5. On a plant-specific basis, more detailed drawings or sketches would be made available by a license renewal applicant if necessary to support NRC review of a license renewal application.			

PWR INTERNALS IR: RESPONSE TO NRC COMMENTS

NRC COMMENT #	NRC COMMENT	INDUSTRY RESPONSE	STATUS	PROPOSED IR REVISION	REMARKS
S-25	Page 4-23, Section 4.8: The interaction effect between thermal aging and irradiation should be addressed in the IR.	Synergistic effects have been addressed in the IRs to the extent that such effects have been observed in actual operation, relevant laboratory data, or related experience. Not all combinations of age-related degradation meet this criterion. The precipitation of chromium-rich delta ferrite and the generation of interstitial vacancies would not seem to be a likely combination that would meet this criterion.	Closed	None	It is believed that the response to this NRC comment fully resolves the concerns raised by the NRC. NUMARC requests that the NRC consider this comment "closed" based upon the response.
S-26	This section indicates that mechanical properties of austenitic stainless steel castings with delta ferrite content below 20% will not be degraded significantly by thermal aging. This statement is not true based on data in NUREG/CR-5385, ANL-89-17 and a paper presented by Slams et al. at SMIRT post conference seminar 6 on Assuring Structural Integrity of Steel Reactor Pressure Boundary Component, August 1983, Monterey, California. Provide justification for the statement.	Following the review of NREG/CR-4513 (ANL-90/42), as well as other information, the IR theshold for potential significance of thermal aging embrittlement to cast austenitic stainless steel (CASS) components was modified to reflect more recent data. The original selection of 20% delta ferrite as the screening threshold was based on a review of exisitng literature, including original IR References 4-29, 4-30, 4-31, 4-32, 4-33. IR Reference 4-33	Open	Section 4.8.2 fo the IR will be revised to reflect this new screening criteria which is now being used consistently through the NUMARC iRs. [p. 4.22, Section 4.82]	It is believed that the response to this NRC comment fully resolves the concerns raised by the NRC. NUMARC requests that the NRC consider this comment "closed" based upon the response.

NRC COMMENT #	NRC COMMENT	INDUSTRY RESPONSE	STATUS	PROPOSED IR REVISION	REMARKS
		compiled a relatively complete set of references covering the period up to its publication in May 1989. Since that time, O.K. Chopra and A. Sather, "Initial Assessment of the Mechanisms and Signifance of Low-Temperature Embrittelement of Cast Stainless Steels in LWR Systems," NUREG/CR-5385 (ANL-89/17) was published in August 1990 and reviewed by the industry. The data was reconciled with the existing results. More recently, O.K. Chopra, "Estimation of Fracture Toughness of Cast Stainless Steels during Thermal Aging in LWR Systems," NUREG/CR-4513 (ANL-90/42), published May 1991, was received. A review of this document provided the information necessary to modify the delta ferrite screening threshold for static-cast components with higher molybdenum content (CF-3M and CF-8M) to 14%. The 20% delta ferrite threshold remains valid for all centrifugally-cast components with lower Mo content (CF-3 and CF-8).			

PWR INTERNALS IR: RESPONSE TO NRC COMMENTS

NRC COMMENT #	NRC COMMENT	INDUSTRY RESPONSE	STATUS	PROPOSED IR REVISION	REMARKS
S-27	Section 4.10 - Fatigue: There is an ongoing generic discussion on fatigue evaluation for license renewal between NUMARC and the staff. Until an agreement(s) is reached, the fatigue issues are unresolved.	Fatigue is an "open" generic issue for the NRC staff. The staff has issued a Branch Technical Position (BTP) on fatigue (issued 12/12/91) which delineates their current thinking on how fatigue should be addressed for license renewal. The industry has agreed that such a process needs to be clearly defined for the license renewal applicant and has offered a logical fatigue reevaluation approach in the Industry Reports. The industry position on fatigue reevaluation and the BTP continue to differ both in form and in substance.	Open	None	Comment S-27 remains "open" for the PWR Internals IR as the general topic of fatigue remains open between the industry and the NRC staff. All other fatigue-related comments for this IR are considered "closed" in that further actions on them are dependent on the resolution of Comment S-27 and the general discussions on fatigue. The fatigue comments considered "closed" are cross-referenced below: - Comment S-4 on p. 1-26 fatigue evaluation discussions. - Comment S-31 on the Section 5.1 Evaluation Method. - Comment S-32 on PWR environment and fatigue crack growth rates. - Comment S-33 on Section 5.1.2.2 Fatigue Re-evaluation. - Comment S-34 on Section 5.1.2.3 Fatigue Crack Growth Evaluation. - Comment S-36 on Section 5.2 Internals Fatigue Evaluations.

PWR INTERNALS IR: RESPONSE TO NRC COMMENTS

NRC COMMENT #	NRC COMMENT	INDUSTRY RESPONSE	STATUS	PROPOSED IR REVISION	REMARKS
S-28	Page 4-31: The IR should provide technical information in support of the newly calculated usage factors.	The summary discussion of the fatigue reanalysis performed by Babcock and Wilcox for their internals, on Page 4-27 through 4-29 of the report, is supported by IR References 4-3, 4-44 and 4-45. The most directly relevant reference for the recalculated fatigue usage factors is IR Reference 4-3, which was provided to the staff at the 4-15-92 NRC/NUMARC PWR internals IR meeting.	Closed	As identified on Page 4-31 of the IR, the supporting technical information for the newly calculated usage factors is contained in Reference 4-3, Project Topical Report: Oconee Nuclear Power Station Unit 1 Reactor Vessel Internals Life Extension Project. The reference document number provided in the IR was Draft EPRI Report RP 2643-18. An alternate reference document is the original report written by Babcock and Wilcox, BAW-2060, October 1988. Both documents are identical. The IR reference document number, for Reference 4-3, will be revised to BAW-2060, October 1988, in place of the EPRI draft document number. [p. 4-27, Section 4.10.2]	It is believed that the response to this NRC comment fully resolves the concerns raised by the NRC. NUMARC requests that the NRC consider this comment "closed" based upon the response. This is the same data which industry shared with the NRC during the fatigue discussions on November 14, 1990.
S-29	Page 4-36, Table 4-1(a): Note 2 should be applied to lower support columns instead of fuel pin.	The reviewer is correct. This was a typo and the IR will be corrected.	Closed	Table was eliminated from IR.	It is believed that the response to this NRC comment fully resolves the concerns raised by the NRC. NUMARC requests that the NRC consider this comment "closed" based upon the response.

PWR INTERNALS IR: RESPONSE TO NRC COMMENTS

NRC COMMENT #	NRC COMMENT	INDUSTRY RESPONSE	STATUS	PROPOSED IR REVISION	REMARKS
S-30	Page 4-36 and 4-27: The IR should clarify that absence of an "S" (i.e., a potentially significant aging issue) in the degradation Table should not eliminate future ISI of that component for that specific form of degradation. In addition, radiation embrittlement should be added in this Table.	We cannot agree. For items that do not contain an "S", no additional ISI is stipulated by the IR beyond that required by current codes, standards, regulations and other commitments made by the license renewal applicant. There is no intent on the part of the IR to supersede these current commitments; instead, the IR limits the scope of license renewal assessment relative to these commitments. Neutron irradiation embrittlement determinations have been included in the revised IR.	Closed	Revise component/degradation matrices to show disposition of all component/degradation mechanism combinations, including neutron irradiation embrittlement.	It is believed that the response to this NRC comment fully resolves the concerns raised by the NRC. NUMARC requests that the NRC consider this comment "closed" based upon the response.
S-31	Section 5.1 Evaluation Method: This section appears to imply that a fracture mechanics analysis may be used in lieu of fatigue usage factor analysis. We do not agree with the alternate method, because of uncertainties in flaw growth predictions. The ASME Code recognizes this limitation and recommends that these flaws should receive augmented inspection in accordance with IWB-2420 of ASME Code Section XI. The IR should be revised to comply with the ASME Code or additional justification is needed.	The issue of effective programs for managing potentially significant fatigue damage has been the subject of ongoing discussions between NUMARC/industry and the staff. The industry position, as articulated in a meeting with the staff on July 16, 1991, is that at least two equivalent paths for evaluating and managing fatigue damage have been demonstrated to be effective -- fatigue analysis/reanalysis, using the procedures of the ASME Code Section III (in this case, Subsection NG), or inservice examination, using the procedures of the ASME Code Section XI (in this case, Subsection IWB).	Closed	Section 5.7 will be modified to discuss fatigue management techniques available to manage potentially significant fatigue-critical components. The use of flaw tolerance (fracture mechanics) analysis will be discussed as a plant-specific alternative in Section 6. [p. 5-12, Section 5-7, p. 6-3, Section 6.2.2]	Comment S-31 is dated based on the NRC staff's Branch Technical Postion (BTP) on fatigue which was issued 12/13/91, after these comments were developed. The BTP more accurately reflects the staff's current thinking on fatigue, including the acceptability fo fatigue reanalysis during the current license term and the use of ASME Section XI inservice inspection/flaw tolerance evaluation as an alternate evaluation method should accurate transient reconstruction not be possible.

NRC COMMENT #	NRC COMMENT	INDUSTRY RESPONSE	STATUS	PROPOSED IR REVISION	REMARKS
		Since the applicable inservice examination requirements for reactor internals (see Table IWB-2500-1, Examination Category B-N-3) emphasize visual (VT-3) examination of accessible surfaces of components that have been removed from the reactor vessel, the industry position is that it would be prudent to supplement the inservice examination alternative with flaw tolerance evaluations, following the general outline of the evaluation steps in ASME Nuclear Code Case N-481. These steps include the postulation of a reference flaw that would be assured of detection, the location of this reference flaw in the worst position and orientation relative to applied stresses, and adequate demonstration that flaw growth and flaw instability do not compromise component integrity. Material properties used in the evaluation would have to be justified by the applicant on a component-specific basis. Therefore, it is not intended that fracture mechanics analysis alone is an alternative to fatigue usage factor evaluation; instead, the flaw tolerance (fracture mechanics) procedures are intended to supplement visual inservice examination requirement.			In agreement between NUMARC and the NRC staff at the 4-15-92 NRC/NUMARC PWR Internals IR meeting, Comment S-31 is "closed" for this IR. Fatigue remains an "open" generic issue as noted in S-27.

PWR INTERNALS IR: RESPONSE TO NRC COMMENTS

NRC COMMENT #	NRC COMMENT	INDUSTRY RESPONSE	STATUS	PROPOSED IR REVISION	REMARKS
S-32	Pages 5-6 through 5-11 and 6-13: (a) The role of the PWR environment including irradiation on fatigue crack growth was apparently not evaluated and should be considered in the IR. (b) The IR states on Page 5-11 that the scaling parameter Co in the crack growth rate formula is not substantially impacted by the PWR water environment. Since according to ASME PVP 1986 Code, the value of Co in the PWR environment could be twice that in air, the IR should be more specific about its recommended value of Co.	The IR will be modified to discuss further the role of PWR water enviroments on fatigue crack growth rates.	Closed	The following will be added to the last paragraph of Section 6.2.2: While estimates of the influence of PWR and BWR water environment effects on stainless steel base and weld metal have been made, ranging from a factor of 2 to 10 higher in crack growth rate than rates in air, the values used in license renewal applications should be justified on a plant-specific basis. For most situations, fatigue crack growth rates in PWR water environments are expected to be minimal. Similarly, it has been demonstrated [6-4] that fast neutron irradiation has no significant effect on the fatigue crack growth of austenitic stainless steels for fluence levels of interest here." Reference: [6-4] L.A. James, "Fatigue-Crack Propagation in Austenitic Stainless Steels, "Atomic Energy Review, Vol. 14, No.1, pp 37-86, 1976.	In agreement between NUMARC and the NRC staff at the 4-15-92 NRC/NUMARC PWR Internals IR meeting, comment S-32 is "closed" for this IR. Fatigue remains an "open" issue as noted in comment S-27.

PWR INTERNALS IR: RESPONSE TO NRC COMMENTS

NRC COMMENT #	NRC COMMENT	INDUSTRY RESPONSE	STATUS	PROPOSED IR REVISION	REMARKS
S-33	Section 5.1.2.2 - Fatigue Re-evaluation This section cites Reference 5-3 as an example of the type of approach that could be used to re-evaluate certain fatigue analyses by using less conservative assumptions than those used in the original analyses. Reference 5-3 is the same as Reference 4-3 which is one of the documents requested in comment No. 25(a) above. There is also ongoing generic discussion on the fatigue evaluation between NUMARC and the staff. Until an agreement is reached, this issue will remain unresolved.	Reference 5-3, which is identical to Reference 4-3, was provided to the staff as indicated in the response to Comment S-28.	Closed	None	In agreement between NUMARC and the NRC staff at the 4-15-92 NRC/NUMARC PWR Internals IR meeting, Comment 5-33 is "closed" for this this IR. Fatigue remains an "open" generic issue as noted in Comment S-27.
S-34	Section 5.1.2.3 Fatigue Crack Growth Evaluation: This section indicates that fatigue crack growth may be calculated based on a scaling parameter for stainless steel in air. This section also indicates that the PWR water environment does not substantially impact the scaling factor. NUMARC should provide more information including test data to support this statement.	The ASME Code Section XI, Subsection IWB, Appendix C and, in particular, Fig. C-3210-1, show that R ratios and temperatures can be accommodated by scaling of the coefficient C_O. The effects of PWR water environment would be expected to be treated in a similar manner, except that the exponent may be affected, as well (see Fig. A-4300-2). However, the justification of fatigue crack growth rates should be a plant-specific consideration in the absence of ASME Code guidance. We believe the response to comment S-32 also addresses this comment.	Closed	See the proposed revision to Comment S-32.	In agreement between NUMARC and the NRC staff at the 4-15-92 NRC/NUMARC PWR Internals IR meeting, Comment 5-33 is "closed" for this this IR. Fatigue remains an "open" generic issue as noted in Comment S-27.

PWR INTERNALS IR: RESPONSE TO NRC COMMENTS

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S-35	Page 5-16: The mechanically redundant systems is discussed in Section 5.1.3 instead of Section 5.1.2.	We agree. This typographical error will be corrected	Closed	Mechanical redundancy is discussed in Section 6.2.1 of the revised IR.	It is believed that the response to this NRC comment fully resolves the concerns raised by the NRC. NUMARC requests that the NRC consider this comment "closed" based upon the response.

PWR INTERNALS IR: RESPONSE TO NRC COMMENTS

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S-36	Section 5.2 - Internals Fatigue Evaluations This section identifies the following two established methodologies that are available to verify the adequacy of the fatigue design basis throughout the license renewal term: (a) Reanalysis in accordance with the procedures described in Section 5.1.2.2; or (b) Inservice Inspection (ISI) in accordance with ASME Section XI. As mentioned in comment No. 31 above, the staff has not yet endorsed Option 1. With respect to Option 2, the staff does not agree that ISI in accordance with the provisions of ASME Section XI is a viable alternative to plant specific analysis which calculates cumulative usage factors. Inspection cannot substitute for analyses, e.g. ISI will not detect fatigue effects that had not yet resulted in crack initiation. Once a crack initiates, rapid propagation may lead to failure between ISI intervals. Historically, ISI has not detected reactor coolant pressure boundary failures before they have occurred. This section should be revised to reflect any agreement(s) to be reached in the ongoing discussion on fatigue evaluation.	We cannot agree with the reviewer's comments. There is no prohibition against fatigue reanalysis during the current license term, provided that such analysis is carried out in accordance with ASME Code Section III procedures. Such re-analysis are performed on a routine basis by current licenses. There is no reason to suspect that fatigue reanalysis is unacceptable for the license renewal term. ISI is not required to detect fatigue crack initiation, just the existence of macroscopic fatigue cracks of a size that approaches acceptance criteria limits of Section XI. The industry recognizes that substitution of ISI for fatigue analysis/reanalysis will not be permitted unless the component location is sufficiently tolerant to such macroscopic flaws to justify the periodic ISI interval. It is the industry's position that ISI has performed well in detecting and sizing flaws that threaten the integrity of the reactor coolant pressure boundary.	Closed	The IR will be revised to reflect agreements reached with the staff of fatigue evaluation.	Comment S-36 is dated based on the NRC staff's Branch Technical Postion (BTP) on fatigue which was issued 12/12/91, after these comments were developed. The BTP more accurately reflects the staff's current thinking on fatigue, including the acceptability fo fatigue reanalysis during the current license term and the use of ASME Section XI inservice inspection/flaw tolerance evaluation as an alternate evaluation method should accurate transient reconstruction not be possible. In agreement between NUMARC and the NRC staff at the 4-15-92 NRC/NUMARC PWR Internals IR meeting, Comment S-36 is "closed" for this this IR. Fatigue remains an "open" generic issue as noted in Comment S-27.

NRC COMMENT #	NRC COMMENT	INDUSTRY RESPONSE	STATUS	PROPOSED IR REVISION	REMARKS
S-37	Section 5.4 - Internals Stress Relaxation Evaluations This section indicates that stress relaxation is not a significant age-related degradation mechanism for the components listed on page 5-18 because of redundancy and periodic visual inspection. It is likely that thermal and irradiation conditions would affect the preload of all bolts or tie rods in the components. Hence, redundancy is not an acceptable justification. Furthermore, visual inspection is not designed to detect stress relaxation. Since many of these components are subject to irradiation embrittlement and stress corrosion cracking, they may be susceptible to multiple and rapid failure. Therefore, all the components listed on page 5-18 need age-related degradation management.	The issue of common-mode failure was addressed in the response to Comment G-20. Thermal and radiation conditions are not likely to affect the pre-load of all bolts or tie rods in the components identically, since the response of each bolt or tie rod depends on chemistry and mechanical factors that vary, in some cases considerably. The variation in bolt hardness and bolt preload alone is sufficient to preclude any common mode failure concerns, unless the loading is extremely severe. The VT-3 visual inspection requirement of the ASME Code Section XI (see IWB-3520.2, e.g.) lists "loose, missing, cracked, or fractured parts, bolting, or fasteners". With consideration of preload variability and other variation taken into account, the relevant conditions of IWB-3520.2 are sufficient to detect and manage potentially significant stress relaxation. We do not agree that the failures will be multiple and rapid.	Open	See the proposed IR revision to Comment G-7.	It is believed that the response to this NRC comment fully resolves the concerns raised by the NRC. NUMARC requests that the NRC consider this comment "closed" based upon the response.

NRC COMMENT #	NRC COMMENT	INDUSTRY RESPONSE	STATUS	PROPOSED IR REVISION	REMARKS
S-38	Section 5.4 - Internals Stress Relaxation Evaluations In this section, it is concluded that all bolts and fuel pins with the exception of Baffle-Former and Core Shroud Assembly Bolts are eliminated from further evaluations based on the following: (a) Bolts and pins represent mechanically redundant systems. (b) Inservice inspection will detect damage and retightening or replacement of the bolts and pins that are found to be damaged is a proven technology. It is not clear how the above conclusions apply to fuel pins. Provide more detailed information to justify the elimination of fuel pins from further evaluation.	The fuel pins referred to are those attached to the core support plate to restrain lateral motion of the fuel assemblies. There is more than one fuel pin per fuel assembly; consequently, there is redundancy. Furthermore, fuel pins are accessible components and any stress relaxation sufficient to cause loosening will be evident by fret marks. Any fuel pins which are damaged will be even more readily detectable. Hence, the IR will use visual inspection as an established program for the evaluation of fuel pins. See also the response to NRC Comment G-6 relative to stress relaxation detection.	Open	Even though redundancy exists, credit will be taken in the IR for inservice inspection to detect fretting, wear or damage to fuel pins. [p. 5-9, Section 5.5] Revise the IR to close the stress relaxation of the fuel pins in Section 5 based upon visual inspection requirements.	It is believed that the response to this NRC comment fully resolves the concerns raised by the NRC. NUMARC requests that the NRC consider this comment "closed" based upon the response.
S-39	Page 5-19, last paragraph: The IR states that wear between two surfaces subject to relative motion is readily detectable by visual inspection long before the effects of wear begin to comprise the structural integrity or function of the component. The current practice of inspection once every 10 years does not make wear readily detectable long before its effects begin to compromise its integrity. The IR should consider an augmented inspection program for all susceptible safety-related components as a requirement for license renewal.	Current ASME Code Section XI inspection requirements (e.g., examination Category B-N-3) are effective in detecting wear on a timely basis. However, such requirements have evolved through the consensus process over the years, and have been evaluated for any time dependencies that might preclude their effectiveness for either the current of the license renewal term. Wear of surfaces is listed as a relevant condition for Examination Category B-N-3 in IWB-3520.2. Industry has determined that this established program is currently effective, and we can find no reason to suspect that the incidence of wear will increase during the license renewal period.	Closed	Section 5.5 of the IR will be revised to discuss the relevant conditions of the VT-3 inspection required by Examination Category B-N-3. [p. 5-9, Para. 2]	It is believed that the response to this NRC comment fully resolves the concerns raised by the NRC. NUMARC requests that the NRC consider this comment "closed" based upon the response.

PWR INTERNALS IR: RESPONSE TO NRC COMMENTS

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S-40	Section 5.6 Internals Thermal Aging Evaluation: This section needs to be rewritten because it indicates thermal aging effects have not been observed in cast austenitic stainless steel components and inservice inspection has not identified any cracking which was attributed to thermal aging. Thermal aging does not cause cracking. it reduces the load carrying capability of cracked components because the material has lower ductility. To ensure that the cast austenitic stainless steel components do not significantly lose their load carrying capability, all the components must be periodically inspected to detect cracks. See General Comment No. 15.	Section 5.6 of the IR will be revised to more accurately describe the effects of potentially significant thermal aging embrittlement on PWR internals, and to describe the elements of effective programs for managing these effects. It is recognized that thermal aging embrittlement of cast austenitic stainless steel components does not, in itself, cause cracking; however, the presence of defects and/or flaws, when combined with the effects of thermal aging embrittlement, is a concern.	Closed	The paragraph at the bottom of p. 5-10 will be revised to read as follows: "Lower support columns and upper support columns in Westinghouse plants, and core support columns in Combustion Engineering plants, are under compressive loads during normal operation. Furthermore, these components are subject only to coolant inlet temperatures which significantly reduces the extent of thermal aging. Cracking caused by thermal aging is unlikely, due to the absence of tensile stress, and the components are subject to limited crack growth because of the presence of compressive stress. Nevertheless, visual inspection conducted at periodic intervals are adequate to manager the effects of potential thermal aging for these support columns. [5-10, Para. 5]	It is believed that the response to this NRC comment fully resolves the concerns raised by the NRC. NUMARC requests that the NRC consider this comment "closed" based upon the response.
S-41	Pages 5-20 and 5-21: Evaluation of thermal aging effects should include the interaction with irradiation where applicable. the IR should consider such interaction especially since visual inspection to detect flaws that may have occurred would not be adequate if slow cracking caused by thermal aging is untimely.	The potential for increased crack growth rates in cast austenitic stainless steel components as the result of a combination of thermal aging embrittlement and neutron irradiation embrittlement has not been identified as a credible synergistic mechanism.	Closed	None	It is believed that the response to this NRC comment fully resolves the concerns raised by the NRC. NUMARC requests that the NRC consider this comment "closed" based upon the response. This comment is similar to Comment S-25.

NRC COMMENT #	NRC COMMENT	INDUSTRY RESPONSE	STATUS	PROPOSED IR REVISION	REMARKS
S-42	Pages 5-20 to 5-23, Section 5-6. The 20 percent ferrite criterion is an inadequate tool to screen cast stainless steel compositions which may be subject to significant embrittlement by thermal aging. An appropriate fracture mechanics analysis is a sound tool for the management of this problem. However, scoping calculations are needed to demonstrate that VT-3 inspections or other inspections are adequate to detect flaws less that the critical size and to ensure that inspection intervals are adequate. As noted previously, we do not believe that VT-3 can reliably detect tight cracks.	The delta ferrite screening thresholds for potential significance of thermal aging embrittlement of cast austenitic stainless steel internals has been discussed extensively in the response to Comment S-26. We agree with the reviewer that a fracture mechanics analysis, such as that described in Section 5.6 of the IR, is effective for management of the potential problem. At issue is the effectiveness of the VT-3 visual inspection to detect cracking prior to reaching a critical size. The industry position is based on the requirements of Examination Category B-N-3, and the definition of relevant conditions cited in IWB-3520.2. In particular, "loose, missing, cracked, or fractured parts" is identified as a relevant condition. The consensus code bodies, in particular the group working on subsection IWG may wish to consider the the use of VT-1. Until such time as the consensus bodies complete their deliberations, the industry is satisfied with detecting cracks with VT-3, followed by supplementary inspection when relevant conditions are found.	Open	None	It is believed that the response to this NRC comment fully resolves the concerns raised by the NRC. NUMARC requests that the NRC consider this comment "closed" based upon the response.

PWR INTERNALS IR: RESPONSE TO NRC COMMENTS

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S-43	Page 6-2: It is not clear from the discussions presented in the IR how the proposed evaluation methods assure detection of stress relaxation.	The report mentions, on Page 6-2, that "A potential also exists for applying UT inspection techniques to bolt stress relaxation, given appropriate baseline data on as-built bolt dimensions." The report does not elaborate, because the procedure discussed is still in the research and development stage. The technique is based on the relatively small changes in propagation velocity of ultrasonic waves along the length of the bolt. If extremely accurate baseline data is available, changes in bolt preload caused by stress relaxation could be detected. The current stage of development of the method does not warrant more detailed discussion.	Closed	None	It is believed that the response to this NRC comment fully resolves the concerns raised by the NRC. NUMARC requests that the NRC consider this comment "closed" based upon the response.