ELECTRIC POWER
RESEARCH INSTITUTE

Materials Reliability Program: Screening, Categorization, and Ranking of Reactor Internals Components for Westinghouse and Combustion Engineering PWR Design (MRP-191)

LICENSED MATERIAL

Technical Report

Effective December 6, 2006, this report has been made publicly available in accordance with Section 734.3(b)(3) and published in accordance with Section 734.7 of the U.S. Export Administration Regulations. As a result of this publication, this report is subject to only copyright protection and does not require any license agreement from EPRI. This notice supersedes the export control restrictions and any proprietary licensed material notices embedded in the document prior to publication.

Materials Reliability Program: Screening, Categorization, and Ranking of Reactor Internals Components for Westinghouse and Combustion Engineering PWR Design (MRP-191)

1013234

Technical Report, November 2006

EPRI Project Manager
H. T. Tang

ELECTRIC POWER RESEARCH INSTITUTE
3420 Hillview Avenue, Palo Alto, California 94304-1338 • PO Box 10412, Palo Alto, California 94303-0813 • USA
800.313.3774 • 650.855.2121 • askepri@epri.com • www.epri.com

DISCLAIMER OF WARRANTIES AND LIMITATION OF LIABILITIES

THIS DOCUMENT WAS PREPARED BY THE ORGANIZATION(S) NAMED BELOW AS AN ACCOUNT OF WORK SPONSORED OR COSPONSORED BY THE ELECTRIC POWER RESEARCH INSTITUTE, INC. (EPRI). NEITHER EPRI, ANY MEMBER OF EPRI, ANY COSPONSOR, THE ORGANIZATION(S) BELOW, NOR ANY PERSON ACTING ON BEHALF OF ANY OF THEM:

(A) MAKES ANY WARRANTY OR REPRESENTATION WHATSOEVER, EXPRESS OR IMPLIED, (I) WITH RESPECT TO THE USE OF ANY INFORMATION, APPARATUS, METHOD, PROCESS, OR SIMILAR ITEM DISCLOSED IN THIS DOCUMENT, INCLUDING MERCHANTABILITY AND FITNESS FOR A PARTICULAR PURPOSE, OR (II) THAT SUCH USE DOES NOT INFRINGE ON OR INTERFERE WITH PRIVATELY OWNED RIGHTS, INCLUDING ANY PARTY'S INTELLECTUAL PROPERTY, OR (III) THAT THIS DOCUMENT IS SUITABLE TO ANY PARTICULAR USER'S CIRCUMSTANCE; OR

(B) ASSUMES RESPONSIBILITY FOR ANY DAMAGES OR OTHER LIABILITY WHATSOEVER (INCLUDING ANY CONSEQUENTIAL DAMAGES, EVEN IF EPRI OR ANY EPRI REPRESENTATIVE HAS BEEN ADVISED OF THE POSSIBILITY OF SUCH DAMAGES) RESULTING FROM YOUR SELECTION OR USE OF THIS DOCUMENT OR ANY INFORMATION, APPARATUS, METHOD, PROCESS, OR SIMILAR ITEM DISCLOSED IN THIS DOCUMENT.

ORGANIZATION(S) THAT PREPARED THIS DOCUMENT

Westinghouse Electric Company LLC

NOTICE:	THIS REPORT CONTAINS PROPRIETARY INFORMATION THAT IS THE INTELLECTUAL PROPERTY OF EPRI. ACCORDINGLY, IT IS AVAILABLE ONLY UNDER LICENSE FROM EPRI AND MAY NOT BE REPRODUCED OR DISCLOSED, WHOLLY OR IN PART, BY ANY LICENSEE TO ANY OTHER PERSON OR ORGANIZATION.

NOTE

For further information about EPRI, call the EPRI Customer Assistance Center at 800.313.3774 or e-mail askepri@epri.com.

Electric Power Research Institute and EPRI are registered service marks of the Electric Power Research Institute, Inc.

Copyright © 2006 Electric Power Research Institute, Inc. All rights reserved.

CITATIONS

This report was prepared by

Westinghouse Electric Company LLC
Nuclear Services
P.O. Box 355
Pittsburgh, Pennsylvania 15230-0355

Principal Investigators
T. Meyer
C. Boggess
S. Byrne
R. Schwirian
F. Gift
R. Gold

This report describes research sponsored by the Electric Power Research Institute (EPRI).

The report is a corporate document that should be cited in the literature in the following manner:

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

REPORT SUMMARY

This report describes the process and results of categorizing Westinghouse and Combustion Engineering (CE) designed pressurized water reactor (PWR) internals components according to age-related degradation and significance. These results are a key element in developing technically sound inspection and evaluation guidelines for.aging management of PWR internals.

Background

Management of aging effects—such as loss of material, reduction in fracture toughness, or cracking—depends on the demonstrated capability to detect, evaluate, and potentially correct conditions that could affect system, structure, or component function. A framework for implementing an aging management program for PWR internals component items and using inspections and flaw tolerance evaluations to manage age-related degradation issues has been developed and is documented in MRP-134 and MRP-153. The important elements of this framework are

- screening, categorizing, and ranking PWR internals component items for susceptibility and significance to age-related degradation mechanisms and
- analyzing and assessing the functionality and safety of PWR internals components to define a safe and cost-effective aging management in-service inspection and evaluation method and strategy.

Initial screening and categorization followed a set of screening criteria defined in MRP-175. A failure modes, effects, and criticality analysis (FMECA) was performed to further screen and categorize internals components based on failure likelihood (susceptibility) and damage likelihood (consequence severity). This report documents the categorization process and results and will be updated following functionality analyses and evaluations.

Objectives

To screen, categorize, and rank Westinghouse- and CE-designed PWR internals components based on susceptibility (failure likelihood) and significance (damage likelihood) to age-related degradation mechanisms in support of developing PWR internals inspection and evaluation guidelines.

Approach

The principal investigators first summarized the Westinghouse- and CE-designed PWR internals components and available screening parameter values such as material, stress, temperature, and neutron exposure. Subsequently, an initial screening of component items to potential age-related

degradation was performed based on MRP-175 screening criteria. Initial screening results were used as inputs to FMECA for further screening and categorization based on failure and damage significance, including financial impact. Finally, component items were binned into three main categories (Category A, Category B, and Category C), defined in MRP-134.

Results

Following the categorization process detailed in this report, the majority of Westinghouse- and CE-designed PWR internals components fall into categories of little and moderate failure likelihood/damage consequence to the aging degradation mechanisms (Categories A and B). For the Westinghouse designs, 13 out of 120 components are of high damage likelihood/consequence (Category C); for the CE designs, the numbers are 6 out of 79. Continued functionality analyses will further categorize components with a more quantitative basis to support the inspection and evaluation (I&E) guidelines development.

EPRI Perspective

The EPRI MRP RI-FG (Materials Reliability Program Reactor Internals Focus Group) has been conducting studies to develop technical bases to support aging management of PWR internals (B&W, Westinghouse, and CE designs), with attention to utility license renewal commitments. This component screening and categorization report is one of a series of reports prepared to provide a basis for developing PWR internals I&E guidelines for utility applications. Related EPRI reports include *Materials Reliability Program: Screening, Categorization, and Ranking of B&W-Designed PWR Internals Component Items (MRP-189)* (EPRI report 1013232, September 2006); *Material Reliability Program: Failure Modes, Effects, and Criticality Analysis for B&W-Designed PWR Internals (MRP-190)* (1013233, September 2006); *Materials Reliability Program: Framework and Strategies for Managing Aging Effects in PWR Internals (MRP-134)* (1008203, June 2005); *Materials Reliability Program: Inspection and Flaw Evaluation Strategies for Managing Aging Effects in PWR Internals (MRP-153)* (1012082, December 2005); and *Materials Reliability Program: PWR Internals Material Aging Degradation Mechanism Screening and Threshold Values (MRP-175)* (1012081, December 2005).

Keywords

PWR internals
Westinghouse design
CE design
Aging management
License renewal
Degradation mechanism
Screening criteria
Screening
Categorization
FMECA

ACRONYMS AND ABBREVIATIONS

ANO	Arkansas Nuclear One
ARDM	Age Related Degradation Mechanisms
ASME	American Society of Mechanical Engineers
ASTM	American Society for Testing and Materials
B&W	Babcock & Wilcox
BMI	Bottom Mounted Instrumentation
CASS	Cast Austenitic Stainless Steel
CE	Combustion Engineering
CEA	Control Element Assembly
CEDM	Control Element Drive Mechanism
Conseq.	Consequence
CS	Core Support
CSB	Core Support Barrel
CSP	Core Support Plate
CUF	Cumulative Usage Factor
dpa	displacements per atom
EPRI	Electric Power Research Institute
FAP	Fuel Alignment Plate
FMEA	Failure Modes and Effects Analysis
FMECA	Failure Modes, Effects and Criticality Analysis
GSSS	Guide Structure Support System
GT	Guide Tube
H	high
HDR	Hold-down Ring
I & E	Inspection and Evaluation

IASCC	Irradiation Assisted Stress Corrosion Cracking
IC	Irradiation Creep
ICI	In-Core Instrumentation
IE	Irradiation Embrittlement
IMT	Issue Management Table
ISR/IC	Irradiation-Induced Stress Relaxation and Irradiation Creep
ITH	Inverted Top Hat
ksi	One thousand pounds per square inch
L	low
LCP	Lower Core Plate
LOCA	Loss-of-Coolant Accident
LP	Lower Plenum
M	Medium
MeV	Million electron volts
MPa	MegaPascals
MRP	Materials Reliability Program
MWt	Megawatt thermal
n/cm^2	neutrons per square centimeter
NSSS	Nuclear Steam Supply System
OBE	Operating Basis Earthquake
Prob.	Probability
PWR	Pressurized Water Reactor
PWSCC	Primary Water Stress Corrosion Cracking
RCCA	Rod Control Cluster Assembly
Req'd.	required
RI	Reactor Internals
RI FA	Reactor Internals Functionality Analysis
RVLMS	Reactor Vessel Level Monitoring System
SCC	Stress Corrosion Cracking
SFE	Stacking Fault Energy
SONGS	San Onofre Nuclear Generating Station

SR	Stress Relaxation
SS	Stainless Steel
SSE	Safe Shutdown Earthquake
TE	Thermal Embrittlement
U.S. NRC	United States Nuclear Regulatory Commission
UCP	Upper Core Plate
UHI	upper head injection
UI	upper internals
UNS	Unified (alloy) Numbering System
UT	Ultrasonic Testing
VS	Void Swelling
VT, VT-1	Visual Testing
WOG	Westinghouse Owners Group
XL	extended length (core)

CONTENTS

LIST OF FIGURES

LIST OF TABLES

1 INTRODUCTION

The current generation of Pressurized Water Reactor (PWR) plants is approaching the end of their respective licensing periods. As plans evolve for extending plant life, the nuclear power industry in the United States and the U.S. Nuclear Regulatory Commission (NRC) have started the process of developing aging management strategies to assess the impact of such extended operation on the safety and reliability of various reactor components and systems. One such program is the EPRI Materials Reliability Program (MRP) Reactor Internals Functionality Analysis (RI FA) that is being performed by Westinghouse under contract.

Specifically, the objective of this program is to perform reactor internals components functionality analyses for Westinghouse and Combustion Engineering (CE) Nuclear Steam Supply Systems (NSSS) designs. This effort includes screening, categorization and ranking, and analysis to support the development of functionality evaluation and inspection guidelines.

The Functionality Analysis project consists of three phases:

- Phase 1–Screening, Categorization and Ranking of Internals Components
- Phase 2–Functionality Evaluations of Lead Internals Components
- Phase 3–Develop Inspection/Mitigation/Repair/Replacement Strategies

The primary focus of this program is to provide the following:

- Screening of reactor internals components for susceptibility to age-related degradation.
- Identification of potential degradation mechanisms in reactor internals based on industry-supplied screening criteria.
- Categorization and ranking of reactor internals components based on a technical evaluation of materials degradation and functionality (including component consequence-of-failure considerations).
- Generation of the technical bases to evaluate continued operation of internals components during extended plant life, to define a safe and cost-effective aging management strategy (including in-service inspection), and to support continued safe and cost-effective operation through mitigation/repair/replacement decisions for internals components.

This report presents the results of the first activity, Screening, Categorization and Ranking of Internals Components. It draws on previous programs for a wealth of information on the identification and management of reactor internals issues for the Westinghouse and CE designs

included in this evaluation. Typical overviews of the Westinghouse and CE internals are presented in Figures 1-1 and 1-2, respectively.

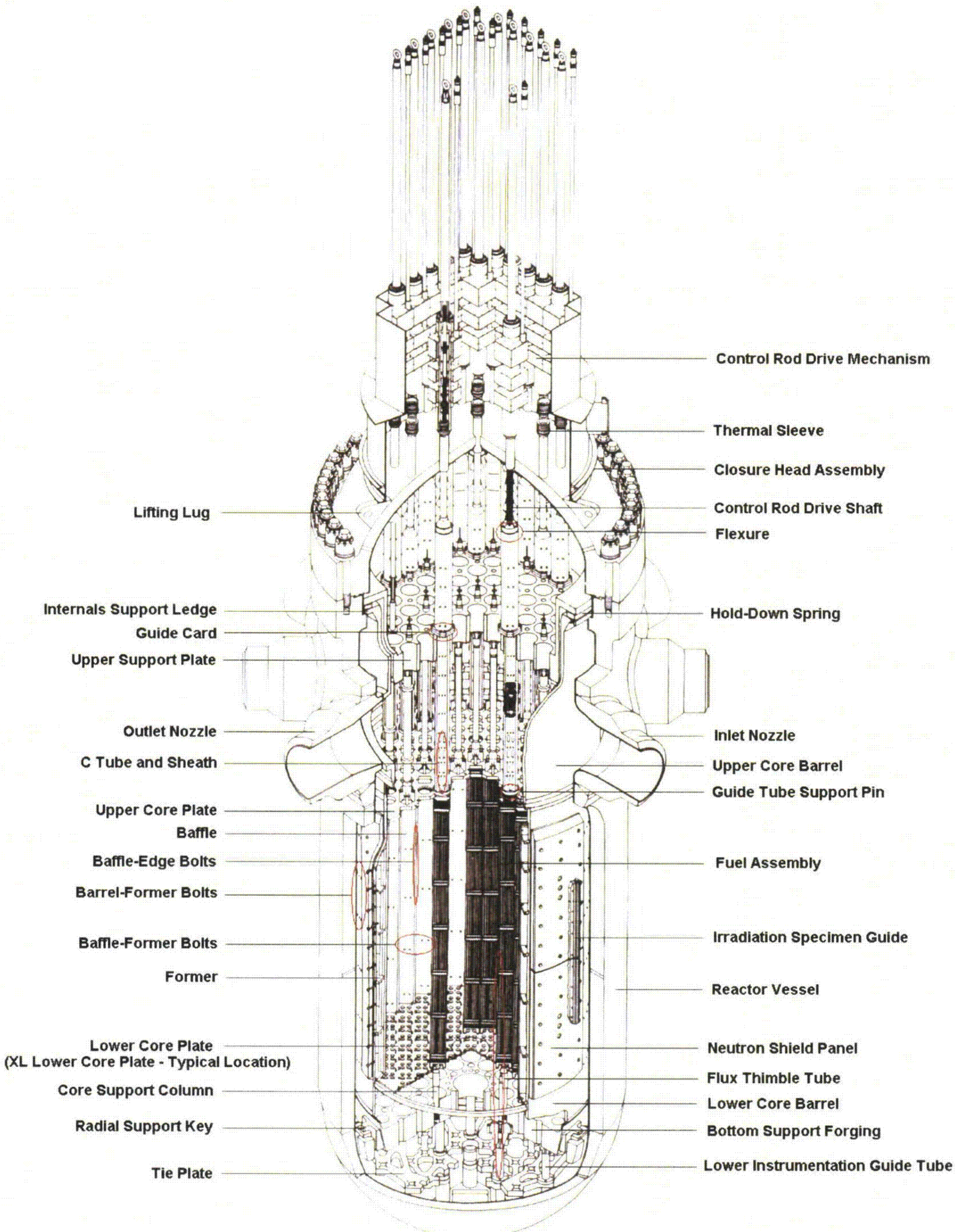


Figure 1-1
Overview of Typical Westinghouse Reactor Internals

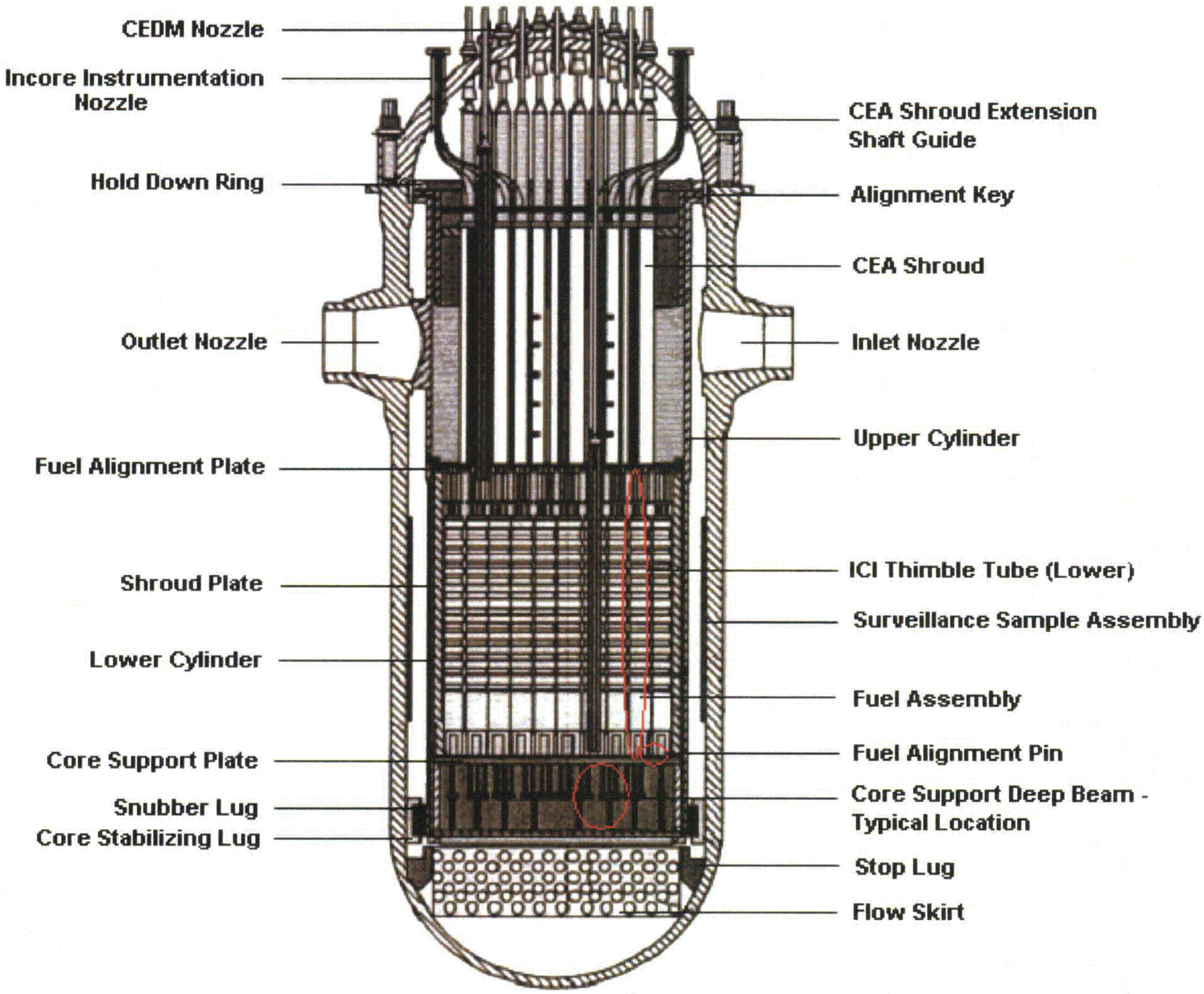


Figure 1-2
Overview of Typical CE Reactor Internals

The effort described in this report consists of the following tasks:

- Identification and Characterization of Internals Components

  The overall project is structured to assess the susceptibility of individual internals components to aging degradation. To achieve this, it was necessary to first identify those individual components and to characterize the conditions under which each component operates. A list of reactor internals components for typical Westinghouse- and CE-designed plants was assembled from the information provided in the Issue Management Tables (IMT) for domestic Pressurized Water Reactors (MRP-156) [1]. Additional component and subcomponent design structure was added as needed to support the project. Information for

each component identified was compiled from available documents and expert interviews to characterize normal operating conditions such as structural loads, temperature, and neutron fluence [2,3].

- Screening Components for Age-Related Degradation Mechanisms

  Based on a comprehensive survey of potential degradation mechanisms in NSSS systems compiled by the MRP, a total of eight degradation mechanisms were identified as potentially relevant to the Westinghouse and CE internals components. Screening criteria for each of these mechanisms were developed and reported in a separate MRP document (MRP-175) [4]. A component-by-component screening assessment was performed to determine which of these degradation mechanisms might threaten the safe and reliable operation of the various components. This process was completed for each of the Westinghouse and CE reactor internals components.

- Failure Modes, Effects and Criticality Analysis (FMECA)

  The screening process identified components that were potentially susceptible to the relevant degradation mechanisms. An expert elicitation process was used by Westinghouse to review each component for the effects of each of the identified degradation mechanism(s), and to assign qualitative estimates of the likelihood for degradation to occur, and the associated likelihood for damage that could result in some degree of functional failure. This procedure generally observed the approach taken in a previous analysis of Westinghouse internals [5]. The FMECA provided the basis for the subsequent categorization and ranking process.

- Categorization and Ranking of Lead Components

  Based on the results of the FMECA evaluation, each component for which a degradation mechanism was concluded to have a relatively high likelihood of occurrence was identified for further evaluation. The potential for such degradation to incur a significant safety risk or unusually high economic penalty was also factored into the selection of components for further evaluation. From this list, Westinghouse identified "lead" components for the subsequent supplementary aging management assessment [6].

The following sections of this report provide detailed descriptions and results of the efforts taken to perform the screening, FMECA, and categorization and ranking process. Section 2 provides a concise outline of the process used. Comprehensive descriptions and reviews of the age-related degradation mechanisms and the screening criteria applied for each are provided in Section 3, followed in Section 4 by the identification, characterization and grouping of components represented in the Westinghouse and CE NSSS designs. Specific application of the screening criteria to the Westinghouse and CE internals components is provided in Section 5. A discussion and summary of the FMECA results is presented in Section 6. The results of the categorization and ranking process identifying all components in Categories A, B, and C are provided in Section 7, followed in Section 8 with overall conclusions.

A number of Appendices are also provided. Appendix A presents a summary of the screening input developed as a product of the expert interview process. Appendix B presents a brief technical discussion of thermal stress relaxation in austenitic stainless steels. Appendices C and D present brief functional descriptions of the internals components of Westinghouse and CE-designed plants, respectively.

2 PROCESS

The EPRI Reactor Internals Functionality Analysis project was structured to initially provide the technical bases for the development of materials aging management programs to mitigate the effects of long-term operation of domestic PWRs. This was facilitated by means of a process to screen, categorize and rank reactor internals components in consideration of the impact of materials degradation mechanisms on the functionality of the individual components. This process included considerations of the potential consequences of failure from a core damage, safety, and financial perspective.

The objective of this section is to provide an outline of the specific process that was applied to the internals components of Westinghouse- and CE-designed PWRs. This process consists of the following tasks:

1. Data Compilation and Review
2. Screening of Components
3. Failure Modes, Effects and Criticality Analysis [FMECA]
4. Categorization and Ranking

The specific input to these tasks included identification of the age-related degradation mechanisms and establishment of the screening criteria values (Section 3), a compilation, characterization and grouping of the internals components for both Westinghouse and CE designs (Section 4) and the initial screening of components with respect to the age-related degradation mechanisms (Section 5). The subsequent tasks encompassed the analysis and documentation of the FMECA (Section 6) which provided the input to the Categorization and Ranking task (Section 7), the conclusions for which are provided in Section 8.

The MRP general framework and strategies are provided in MRP-134 [6]. That document defines a categorization process that leads to the development of Inspection and Evaluation (I&E) guidelines for the internals. Four categories of components for classification of the significance to aging degradation effects (A, B, B' and C) are defined, as shown in the flow diagram reproduced in Figure 2-1.

The definitions for each category described in MRP-134 [6] are as follows:

Category A

Category A components are those for which aging effects are below the screening criteria, so that aging degradation significance is minimal. Typically, only the required ASME B&PV Code Section XI Examination Category B-N-3 ISI visual examinations (VT-3) will be performed on these components to assess potential aging effects.

Category C

Category C PWR internals components are those "lead" components for which aging effects are above screening levels, which have moderate or high susceptibility to degradation, and have not yet been demonstrated to be sufficiently tolerant to remain functional relative to aging degradation significance. Enhanced inspections (e.g., Enhanced VT-1, UT, etc.) and/or surveillance sampling will typically be warranted to assess aging effects and verify functionality of these components.

Category B

Category B includes those PWR internals components that are moderately susceptible to the aging effects, such that the effects on function cannot easily be dispositioned by screening and are not "lead" components. Category B components may require additional evaluations to be shown tolerant of the aging effects with no loss of functionality (i.e., damage tolerant). These components are candidates for a better than VT-3 quality remote visual examination. An example of one such technique, termed VT-3/VT-1, could be implemented; implying that the remote examination may satisfy the standoff distance and character recognition requirements for a VT-1 visual examination.

Category B'

Category B' components are those "lead" components that can be shown to be tolerant of the aging effects through a functionality assessment. Aging degradation significance for these PWR internals components is considered to be manageable using a better than VT-3 quality remote visual examination (e.g., VT-3/VT-1), but they may also be candidates for an expanded inspection program.

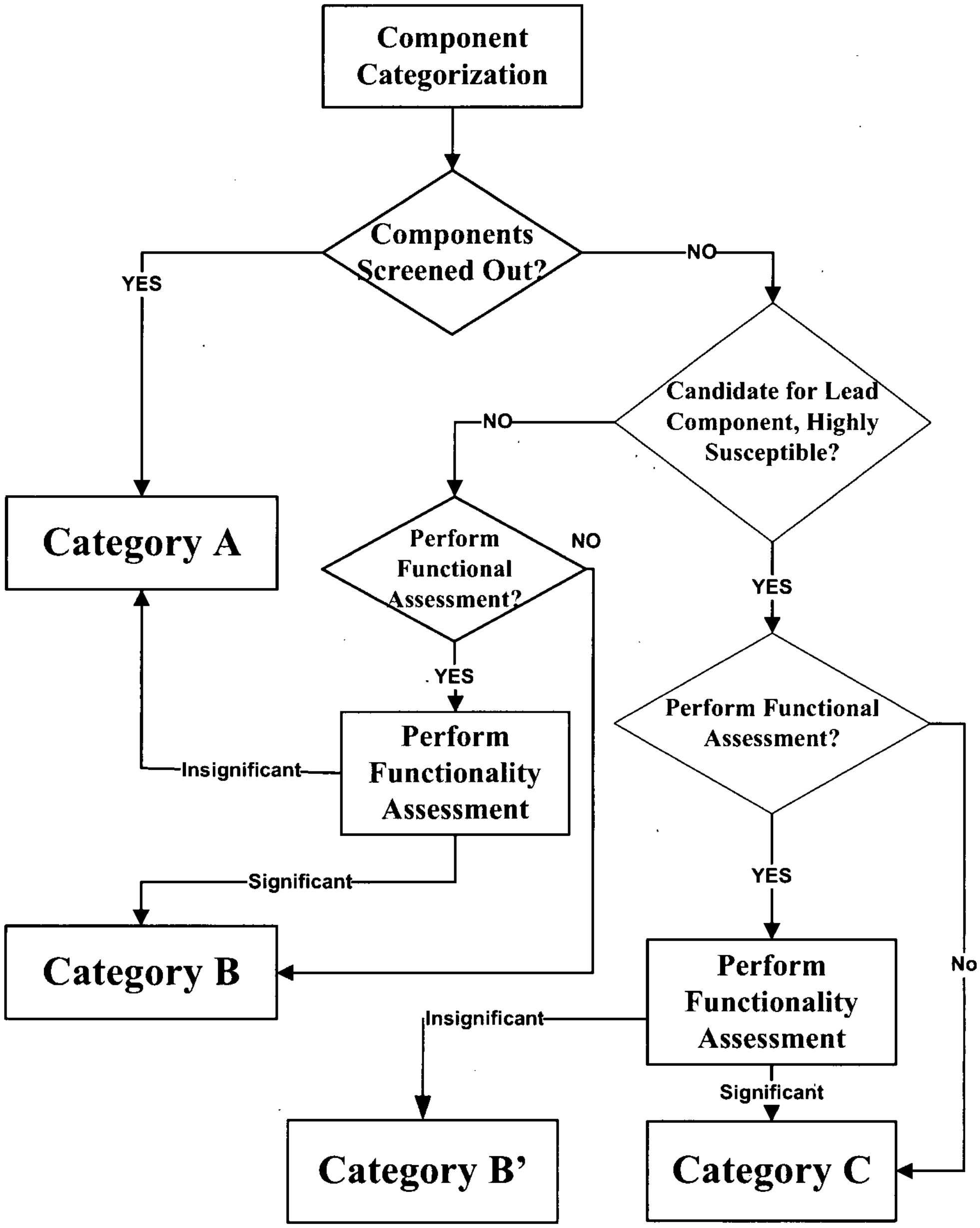


Figure 2-1
Categorization Flow Diagram from MRP-134 [6]

In Figure 2-1, the initial screening step identifies those components that fall below the screening criteria, indicating little susceptibility to the aging degradation mechanisms. These are placed in Category A. The next step requires an evaluation of the remaining components to determine whether or not the component is a candidate for "lead" component designation. If such evaluation concludes this to be the case, the components are placed into the logic path leading to Category C (highest susceptibility to aging effects).

The remaining components not assigned to Category C are placed into the logic path leading to Category B (moderately susceptible to aging effects). Depending on the judgment of the expert panel performing these assessments, these latter components may be immediately identified as Category B or Category A or they may require a functionality assessment, the results of which can assign the components to Category A.

Returning to the Category C path, key components are selected for more detailed functionality assessments. If the results of the functionality assessment demonstrate that the component is tolerant of the expected aging degradation, it may be moved to Category B' (damage tolerant).

Westinghouse elaborated and implemented the MRP-134 process as illustrated in Figure 2-2 and described below.

2.1 Process Implementation

The effort described in this report consisted of the screening, FMECA, and ranking and categorization of the reactor internals components as shown in Figure 2-2. These activities are shown within dashed brackets of the figure. The Functionality Analyses are not represented explicitly in Figure 2-2.

The specific actions taken pursuant to these tasks were:

- Data Compilation and Review

  The individual components and the conditions under which each component operates were characterized. A list of reactor internals components for typical Westinghouse and CE-designed PWRs was assembled from information provided in the Issue Management Table (IMT) for domestic PWRs and other available documents [1,3]. See Appendices C and D for a summary description of the internals components from the IMT [1]. Additional information was solicited from an expert panel and used to characterize structural loads, temperatures, and the radiation environment. This information was assembled into a database (see Tables A-1 and A-2) and was reviewed for accuracy and completeness.

- Screening of Components

  The degradation mechanisms and the screening criteria used for this task were taken from previously published assessments [4]. The screening criteria were applied to the component materials database described in Section 4 to screen all components for susceptibility to each of the age-related degradation mechanisms (ARDM).

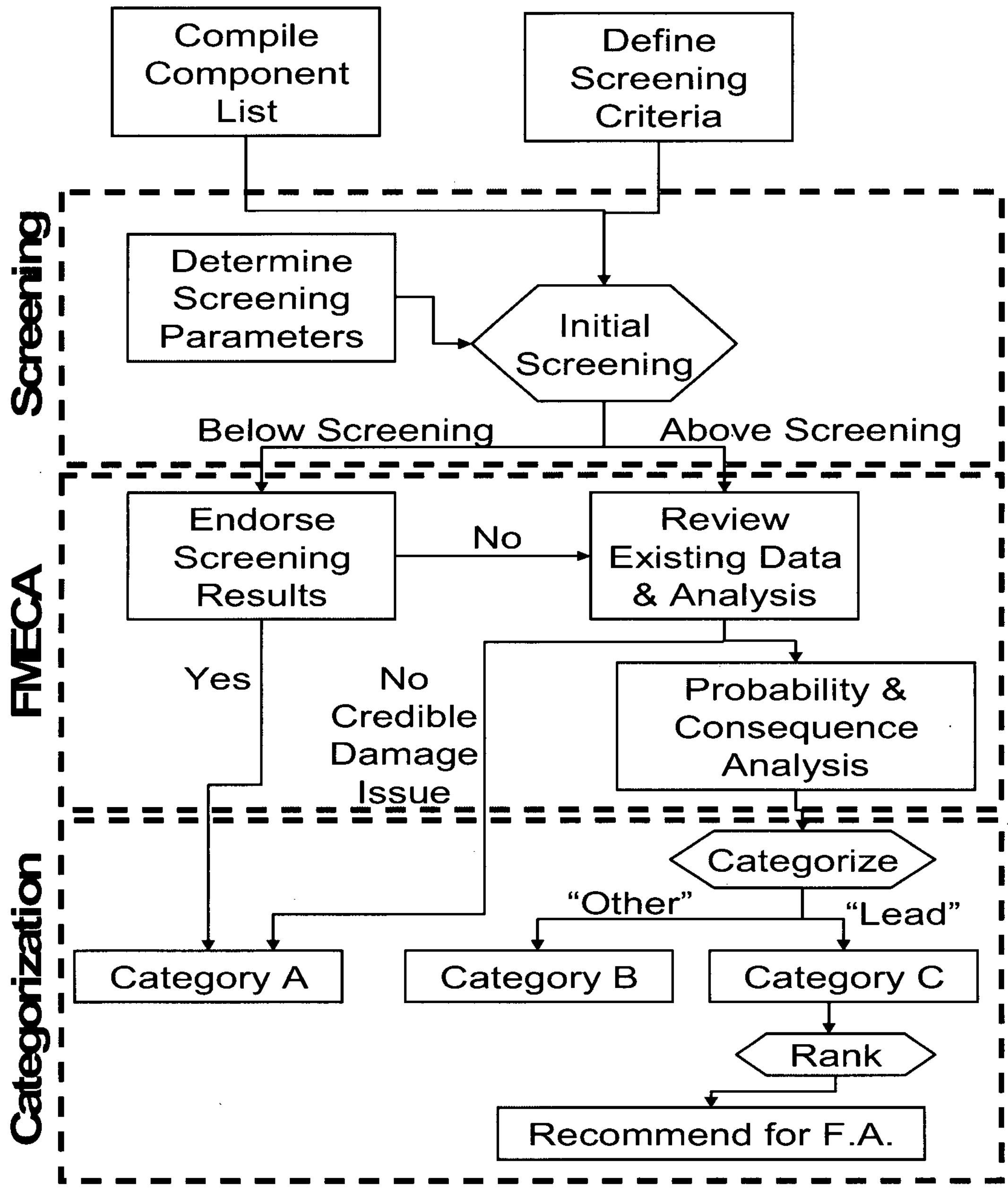


Figure 2-2
Implementation of Screening, FMECA and Categorization Process

- Failure Modes, Effects and Criticality Analysis (FMECA)

An evaluation team consisting of knowledgeable Westinghouse individuals was assembled as an Expert Panel to provide expertise in the areas of reactor internals designs, materials age-related degradation mechanisms, safety analysis, and asset management. The expert panel reviewed the results of the susceptibility screening and endorsed or modified the results. The final results of this assessment were used in a "nine-box assessment" (3x3 matrix

with low, medium or high probabilities of degradation vs. low, medium or high probabilities of damage/plant reliability) in order to arrive at a preliminary grouping of components for the categorization and ranking task.

- Categorization and Ranking

  The expert panel used the initial ranking of component/effects to assign each screened-in component to one of the initial categorizations (Category A, B or C). Factors that were considered in this activity included: the probability of occurrence of a degradation mechanism, the probability of failure, and the severity of any consequences with respect to operation, safety, financial impact and plant reliability. The Category C components were identified for consideration as "lead" components for the functionality analyses.

3 AGE RELATED DEGRADATION MECHANISMS AND SCREENING CRITERIA

The age-related degradation mechanisms (ARDM) and screening criteria used were those defined in MRP-175 [4,7].

Eight potential age-related degradation mechanisms for reactor internals were identified:

1. Stress Corrosion Cracking (SCC)
2. Irradiation Assisted Stress Corrosion Cracking (IASCC)
3. Wear
4. Fatigue
5. Thermal Aging Embrittlement
6. Irradiation Embrittlement
7. Void Swelling
8. Thermal and Irradiation-induced Stress Relaxation or Irradiation Creep

The screening criteria set thresholds for evaluating each degradation mechanism. Each of the ARDMs and the associated screening criteria are summarized in the following sections. The Westinghouse application of the screening criteria is described in Section 5.

3.1 Stress Corrosion Cracking (SCC)

Stress corrosion cracking (SCC) refers to the environmental degradation of structural materials as a response to aggressive combinations of stress, environment (including temperature) and low material resistance to the corrosion process of interest. In PWRs a common form of SCC is the intergranular stress corrosion cracking of austenitic Ni-base alloys such as Alloy 600 and Alloy X-750 in primary water environments; this is referred to as PWSCC. In general, PWR water has an extremely low oxidation potential and SCC is not a concern for most alloys, including the extensively used austenitic stainless steels. Although there is very little evidence of stress corrosion cracking in these alloys, occasional occurrences have been reported in the literature and, hence, stress thresholds have been set for each material.

Stainless steel welds with a high degree of constraint, and parts with severe (>20%) cold work were identified as potential contributors to SCC susceptibility of austenitic stainless steels. Small

weld locations such as tack welds or plug welds were excluded in accordance with MRP-175 [4,7], but only after the concurrence of the FMECA Expert Panel. Structural welds used to join reactor internals components were considered more likely than tack welds or plug welds to create a high degree of constraint.

Low ferrite level (<5%) was identified as contributing to SCC susceptibility for cast austenitic stainless steels and stainless steel welds.

Based on laboratory experience, higher stress levels were set for precipitation-hardened stainless steels, precipitation-hardened nickel-base alloys, and martensitic stainless steels.

The SCC criteria from MRP-175 [4,7] are summarized in Table 3-1. [More detailed explanations of the table entries are available in MRP-175.]

Table 3-1
Stress Corrosion Cracking (SCC) Screening Criteria for PWR Internals Materials [4]

Material	Parameter	Value
Austenitic Stainless Steels	Stress and Material	≥ 30 ksi (207 MPa) and Cold-work ≥20% or Welded Locations
Austenitic Stainless Steel Welds	Stress and Material	≥ 30 ksi (207 MPa) and Ferrite < 5%
Martensitic Stainless Steels	Stress	≥ 88 ksi (607 MPa)
Martensitic PH Stainless Steels	Stress	≥ 88 ksi (607 MPa)
Austenitic PH Stainless Steels	Stress and Material	≥ 70 ksi (483 MPa) and Surface cold-work
	Hot-headed or shot-peened bolting that meet the stress criterion are to be evaluated for SCC.	
Cast Austenitic SS	Stress and Material	≥ 35 ksi (241 MPa) and Ferrite < 5%
Austenitic Ni-base Alloys	Stress	≥ 30 ksi (207 MPa)
Austenitic Ni-base Welds	Stress	≥ 35 ksi (241 MPa)
Austenitic PH Ni-base (Alloy X-750)	Stress	≥ 100 ksi (689 MPa)
	AH and BH condition considered more susceptible than HTH condition.	
Austenitic PH Ni-base (Alloy 718)	Stress	≥ 130 ksi (896 MPa)
Co-base Alloys	Alloys not susceptible in PWR internals locations.	

3.2 Irradiation Assisted Stress Corrosion Cracking (IASCC)

IASCC is a unique form of SCC that occurs only in highly irradiated components. While evaluation of IASCC susceptibility requires additional consideration of the neutron exposure, the consequence of this degradation is a stress corrosion crack similar to the cracking observed from SCC.

A fluence-dependent screening stress was established for IASCC in austenitic stainless steels in MRP-175 [4]. Those criteria are given in Table 3-2 and are generally based on the limited experimental curve fit presented in Figure 3-1. A threshold fluence for IASCC occurs at approximately 3 dpa. Above this fluence threshold, the required stress drops sharply with increasing neutron dose.

Table 3-2
Irradiation Assisted Stress Corrosion Cracking (IASCC) Screening Criteria [4]

Material	Parameter	Value
All Alloys	Stress and Dose	See SCC criteria (Table 3-1) (IASCC not considered applicable) and < 2.0 X 10^{21} n/cm^2 (E > 1 MeV) [< 3 dpa]
		≥ 89 ksi (616 MPa) and 2.0 X 10^{21} n/cm^2 (E > 1 MeV) [3 dpa]
		≥ 62 ksi (425 MPa) and 6.7 X 10^{21} n/cm^2 (E > 1 MeV) [10 dpa]
		≥ 46 ksi (315 MPa) and 1.3 X 10^{22} n/cm^2 (E > 1 MeV) [20 dpa]
		≥ 30 ksi (207 MPa) and 2.7 X 10^{22} n/cm^2 (E > 1 MeV) [40 dpa]

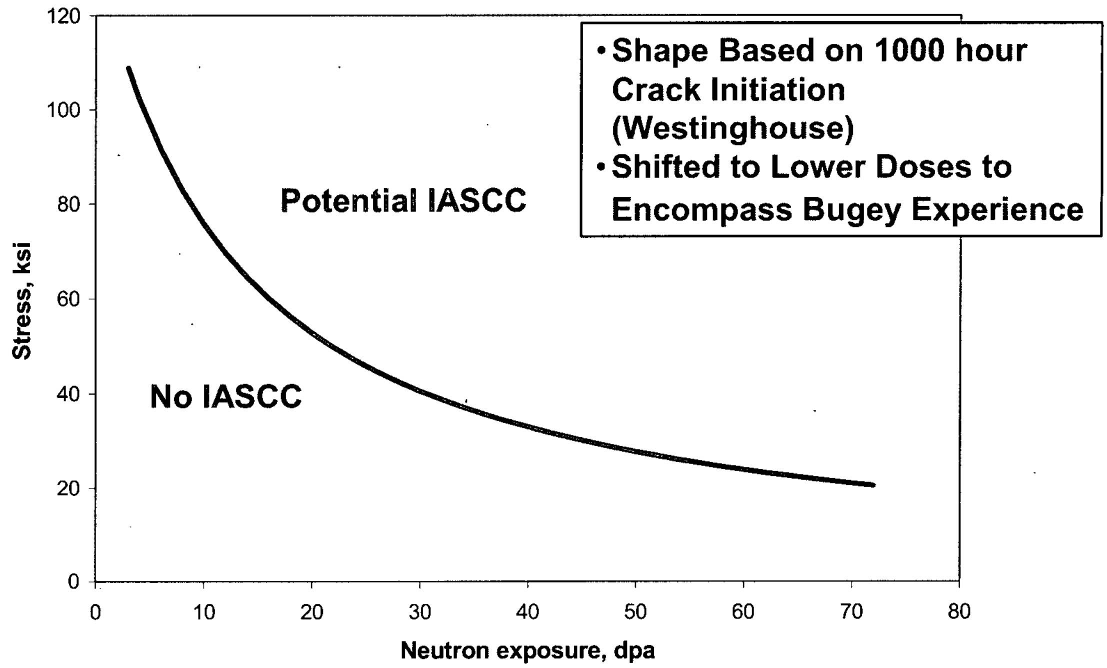


Figure 3-1
MRP-175 Screening Criteria for IASCC [4]

3.3 Wear

The extent to which wear will occur is determined by the relative properties of the two materials, the surface condition and their relative motion. The guidance for screening in Table 3-3 suggests that components should be screened in for wear when two surfaces are in contact and there is a potential for relative motion. As this is a subjective criterion, the stress analysts, reactor internals designers and materials engineers were asked to identify components with the potential for wear. Components subject to stress relaxation that might result in modes of wear that were not originally anticipated were automatically screened in for wear. All components screened in for wear based on the recommendation of the panel were differentiated from those items that were screened in for wear solely on the basis of stress relaxation.

Table 3-3
Wear Screening Criteria [4]

Material	Criteria	
	Parameter	**Locations**
All Alloys	Relative motion	Locations where this may occur between surfaces of adjacent components.
		Example: control rod guide tubes
Material	**Criteria**	
	Parameter	**Locations**
All Alloys	Clamping force	Locations where this is required.
		Example: mating ledge between internals and RV
	Bolted or spring items	Locations where SR/IC is screened as applicable.
		Example: baffle-to-former bolts

3.4 Fatigue

The fatigue screening criteria are outlined in Table 3-4. MRP-175 identifies a cumulative usage factor of 0.1 at the end of the original design life (40 years) as the screening criterion. This relatively low value was chosen to allow for extended operation and to account for the possible effects of environment on fatigue life.

Table 3-4
Fatigue Screening Criteria [4]

Material	Criteria	
	Parameter	Value
All Alloys	CUF	≥ 0.1
	Bolted or spring items	Locations where SR/IC is screened as applicable.
	As material aging concerns with IE, SR/IC, etc. occur, low cycle fatigue and/or high cycle fatigue may become an issue.	
	In some instances fatigue life was alternatively qualified through testing. These component items should be initially screened in for potential fatigue concerns and evaluated.	

3.5 Thermal Aging Embrittlement (TE)

The mechanical properties of some materials are altered by exposure to neutron irradiation or by long term thermal aging. Because these changes often include a loss in tensile ductility or toughness, these phenomena are generally referred to as embrittlement. Increases in yield stress, ultimate stress and notch sensitivity (decreased fracture toughness) are also commonly observed. Whereas SCC can result in the formation of a crack in a component, changes to the strength or ductility do not directly result in cracking or other visible effects in the component. However, those components that must respond to significant loads or require a high flaw tolerance during normal or transient conditions may become more susceptible to cracking due to embrittlement. The screening criteria from MRP-175 define the conditions that can be expected to produce significant changes in the mechanical properties [4]. Further evaluation is required to determine the significance of these changes.

Long term thermal aging can reduce the fracture toughness of martensitic stainless steels. Cast stainless steels with high ferrite contents can also undergo thermal embrittlement due to long term thermal aging. The thermal embrittlement screening criteria screen in all martensitic stainless steels and cast austenitic stainless steels with high ferrite contents as outlined in Table 3-5.

Table 3-5
Thermal Aging Embrittlement (TE) Screening Criteria [4]

<table>
<tr><th rowspan="2">Material</th><th colspan="2">Criteria</th></tr>
<tr><th>Parameter</th><th>Value</th></tr>
<tr><td>Austenitic SS
Austenitic PH SS
Austenitic Ni-Base Alloys
Austenitic PH Ni-Base Alloys
Co-Base Alloys</td><td colspan="2">TE is not applicable to these materials.</td></tr>
<tr><td>Cast Austenitic SS
(Centrifugal Castings)</td><td>Ferrite</td><td>> 20%</td></tr>
<tr><td rowspan="2">Cast Austenitic SS
(Static Castings)</td><td>Molybdenum
and
Ferrite</td><td>≤ 0.50%
and
> 20%</td></tr>
<tr><td>Molybdenum
and
Ferrite</td><td>> 0.50%
and
> 14%</td></tr>
<tr><td rowspan="3">Austenitic SS Welds</td><td>Molybdenum
and
Ferrite</td><td>≤ 0.50%
and
> 20%</td></tr>
<tr><td>Molybdenum
and
Ferrite</td><td>> 0.50%
and
> 14%</td></tr>
<tr><td colspan="2">TE is not anticipated as an issue due to ASME Code procurement requirements for low levels of ferrite (5-15%) and low Mo levels.</td></tr>
<tr><td>Martensitic SS</td><td colspan="2">All component items considered susceptible to TE.</td></tr>
<tr><td>Martensitic PH SS</td><td colspan="2">All component items considered susceptible to TE.</td></tr>
</table>

3.6 Irradiation Embrittlement (IE)

Irradiation-induced embrittlement (IE) in austenitic stainless steels has been the subject of extensive research. Under PWR irradiation conditions, a fast neutron (E > 1 MeV) fluence of $1x10^{21}$ n/cm^2 (~1.5 dpa) will produce a large increase in the tensile yield strength of austenitic stainless steels. Limited data on cast austenitic stainless steels indicate that they behave in a similar fashion. However, it has been suggested that a synergistic effect between thermal and irradiation embrittlement mechanisms may occur in the cast austenitic steels. Therefore, the irradiation embrittlement screening criterion for cast austenitic stainless steels was set more conservatively as $6.7x10^{20}$ n/cm^2 (~1 dpa).

The irradiation embrittlement screening criteria are summarized in Table 3-6.

Table 3-6
Irradiation Embrittlement (IE) Screening Criteria [4]

Material	Criteria	
	Parameter	**Value**
Austenitic PH SS Austenitic Ni-Base Alloys Austenitic PH Ni-Base Alloys Martensitic SS Martensitic PH SS Co-Base Alloys	These materials are used in relatively low fluence locations; therefore, IE is not an applicable age-related degradation mechanism for component items fabricated with these alloys.	
Austenitic SS	Dose	$\geq 1 \times 10^{21}$ n/cm^2 (E > 1 MeV) [$\geq$ 1.5 dpa]
Austenitic SS Welds Cast Austenitic SS	Dose	$\geq 6.7 \times 10^{20}$ n/cm^2 (E > 1 MeV) [$\geq$ 1 dpa]
	Lower screening values used are to account for large initial fracture toughness variability with these materials and possible synergistic effects with thermal aging embrittlement	

3.7 Void Swelling (VS)

The agglomeration of irradiation-induced point defects in highly irradiated steels can produce crystallographic voids in a material. The accumulation of a large number of voids can result in swelling of the affected component. There is typically an incubation exposure required prior to the onset of void swelling (VS). This incubation dose is controlled by a number of variables including neutron fluence, neutron flux (dose rate), neutron spectrum, irradiation temperature, and material condition. The magnitude of void swelling tends to increase with higher neutron fluence and higher temperature and with lower neutron flux. Material variables such as chemical composition and cold work are a factor in a fast breeder reactor environment, but there are no irradiation data under PWR conditions to demonstrate the material dependence. The accumulation of the high neutron fluence needed to produce voids at low neutron fluxes and low irradiation temperatures in PWR internals would require long term exposures. Most of the available data (i.e., those from fast breeder reactors) were obtained after irradiation for several years to over 20 dpa at temperatures above approximately 750°F (~400°C). Neutron flux and irradiation temperature are substantially lower in the PWR reactor internals such that it is expected to take many years to obtain meaningful data.

The screening criteria established for void swelling are a minimum fast fluence of 1.3×10^{22} n/cm^2 (E > 1 MeV) (~20 dpa) and a minimum temperature of 608°F (320°C) (Table 3-7). This minimum temperature is well above the core inlet temperature, meaning that the susceptible locations will be primarily limited to regions closer to the core outlet temperature, or regions with high gamma heating rates.

Table 3-7
Void Swelling (VS) Screening Criteria [4]

Material	Criteria	
	Parameter	Value
Cast Austenitic SS Austenitic PH SS Austenitic Ni-Base Alloys Austenitic PH Ni-Base Alloys Martensitic SS Martensitic PH SS Co-Base Alloys	These materials are used in relatively low temperature and fluence locations; therefore, VS is not an applicable age-related degradation mechanism for component items fabricated with these alloys.	
Austenitic SS Austenitic SS Welds	Temperature and Dose	$\geq$ 608°F (320°C) and $\geq 1.3 \times 10^{22}$ n/cm^2 (E > 1 MeV) [$\geq$ 20 dpa]

3.8 Thermal and Irradiation-induced Stress Relaxation or Irradiation Creep (ISR/IC)

Time-dependent plastic deformation of materials can alter the mechanical response of a system. When a component is maintained at a constant load, the time-dependent plastic deformation is defined as creep. If a constant displacement is imposed on a component, the time-dependent response may decrease the elastic loads and the phenomenon is defined as stress relaxation. These phenomena are thermally activated and it is possible to observe large values of creep or stress relaxation at elevated temperatures. Neutron irradiation can also enhance time-dependent plastic deformation. At significant irradiation damage levels, irradiation-induced creep and irradiation-enhanced stress relaxation can occur at temperatures well below the thermal activation range.

Many bolts and springs require a pre-stress to properly satisfy their function. Stress relaxation in these components may, therefore, be a precursor to failure, in view of the fact that it may reduce the preload sufficiently to create the potential for subsequent wear or fatigue. The stress relaxation screening criteria in Table 3-8 were deemed to apply to all bolts and springs that require pre-stress for functionality and for which the temperature is considered sufficiently high for thermally-activated stress relaxation to occur. In addition, the screening criteria identify all bolts and springs irradiated to neutron fluence values greater than 1.3×10^{20} n/cm^2 (E > 1 MeV) (~0.2 dpa) as potentially subject to irradiation-induced stress relaxation.

Table 3-8
Thermal and Irradiation-Induced Stress Relaxation or Irradiation Creep (ISR/IC) Screening Criteria [4]

Material	Criteria	
	Parameter	**Value**
Thermal SR		
All Alloys	Bolts or springs	All locations
	Applies to component items that require preload for functionality.	
Irradiation-Enhanced SR and IC		
All Alloys	Dose	$\geq 1.3 \times 10^{20}$ n/cm^2 (E > 1 MeV) [$\geq$ 0.2 dpa]
	Applies to all bolted or spring locations. Complex interactions when VS occurs.	

4 IDENTIFICATION, GROUPING AND CHARACTERIZATION OF REACTOR INTERNALS COMPONENTS

4.1 General Identification and Grouping of Westinghouse and Combustion Engineering PWR internals

The Westinghouse fleet consists of 62 plants in the United States. This includes both the Westinghouse designs and the Combustion Engineering (CE) designs as shown in Table 4-1. The large number of plants and designs made it necessary to establish plant groupings to provide a manageable number of designs and operating conditions. In keeping with the overall objective of this program to provide a comprehensive aging management program for the reactor internals, it was necessary to generate a list of components and a list of plant groupings for which operating parameters could be obtained and which were representative of all Westinghouse and CE plants.

Table 4-1
Westinghouse and Combustion Engineering-Designed Plants

Westinghouse-Designed Plants		
RE Ginna	Point Beach 1	HB Robinson 2
Point Beach 2	Turkey Point 3	Surry 1
Surry 2	Turkey Point 4	Prairie Island 1
Kewaunee	Indian Point 2	Prairie Island 2
DC Cook 1	Indian Point 3	Beaver Valley 1
Salem 1	Farley 1	North Anna 1
DC Cook 2	North Anna 2	Farley 2
Sequoyah 1	Salem 2	McGuire 1
Sequoyah 2	VC Summer 1	McGuire 2
Callaway 1	Diablo Canyon 1	Catawba 1
Byron 1	Wolf Creek 1	Diablo Canyon 2

Table 4-1 (continued)
Westinghouse and Combustion Engineering-Designed Plants

Westinghouse-Designed Plants		
Millstone 3	Catawba 2	Shearon Harris
Vogtle 1	Byron 2	Beaver Valley 2
Braidwood 1	South Texas 1	Braidwood 2
Vogtle 2	South Texas 2	Comanche Peak 1
Comanche Peak 2	Watts Bar 1	Seabrook 1
Combustion Engineering-Designed Plants		
Palisades	Fort Calhoun	Calvert Cliffs 1
Calvert Cliffs 2	Millstone 2	St. Lucie 1
ANO-2	SONGS 2	SONGS 3
St. Lucie 2	Waterford 3	Palo Verde 1
Palo Verde 2	Palo Verde 3	

The starting point for determining the list of components to be evaluated was the IMT compiled in MRP-156 [1]. Reactor internals components and the materials of construction were identified in the IMT for both Westinghouse and Combustion Engineering-designed reactors. A standardized nomenclature for reactor components does not exist, so there are many minor variations in component designs that complicate the process of component identification. Compilation of a generalized list of reactor components, therefore, requires a considerable amount of engineering judgment. An industry-appointed panel was responsible for assembling the general reactor internals component lists for the different vendor plant designs in the IMT. The Westinghouse and Combustion Engineering component lists in the IMT were subsequently reviewed and augmented by Westinghouse for evaluation in this program.

To assist in managing this evaluation, plant groupings were established to create a classification scheme (or hierarchy) in which specific Westinghouse and Combustion Engineering-designed plants would be placed. Plant groupings were intended to reduce the complexity of input needed to screen internals components and to simplify a specific plant's application of the final conclusions and recommendations developed through the subsequent Screening, FMECA, and Categorization and Ranking efforts. The highest level division in the plant groupings differentiates the Westinghouse and Combustion Engineering reactor internals designs. The high-level bisection was necessary to identify significant differences in design and nomenclature, operating conditions, and materials of construction that were unique to each of the original vendor designs.

The twenty-one Westinghouse and six Combustion Engineering plant groupings identified in Tables 4-2 and 4-3 were used by the panel to identify features common among different plants. While these groupings do not exhaust the design variations among Westinghouse and

Combustion Engineering plants, these features were used in the application of the available data (within plant-specific analyses) and to provide the necessary input for the screening process. Westinghouse-designed plants were divided into three general categories: 2-loop, 3-loop, and 4-loop plants. CE-designed plants were also divided into three general categories: bolted core shroud with top-mounted in-core instrumentation (ICI), welded core shroud with top-mounted in-core instrumentation, and welded core shroud with bottom-mounted in-core instrumentation.

Subdivisions within these general groups for both designs were added to further distinguish significant variations in the internals structure and/or materials of construction within each category.

Table 4-2
Plant Groupings for Westinghouse-Designed Plants

Plant Group	Plant Name	Original MWt	No. of Loops	Baffle Barrel Region Design
1	Ginna, Kewaunee	1300 1650	2	Downflow
2	Point Beach 1, 2	1518	2	Converted Upflow
3	Prairie Island 1, 2	1650	2	Downflow
4	Turkey Point 3, 4 HB Robinson 2	2200	3	Downflow
5	Surry 1, 2	2441	3	Downflow
6	Beaver Valley 1	2652	3	Converted Upflow
7	North Anna 1	2775	3	Converted Upflow
8	North Anna 2	2775	3	Downflow
9	Farley 1, 2	2652	3	Converted Upflow
10	VC Summer	2775	3	Downflow
11	Shearon Harris	2775	3	Upflow
12	Beaver Valley 2	2652	3	Upflow
13	Cook 1 Indian Point 2, 3	3250 2758 3025	4	Downflow
14	Diablo Canyon 2	3411	4	Downflow
15	Diablo Canyon 1 Salem 1, 2; Cook 2	3411 3391	4	Downflow

Table 4-2 (continued)
Plant Groupings for Westinghouse Designed Plants

Plant Group	Plant Name	Original MWt	No. of Loops	Baffle Barrel Region Design
16	Byron 1, 2 Braidwood 1, 2	3411	4	Upflow
17	Callaway Wolf Creek Comanche Peak 1, 2 Seabrook Vogtle 1, 2 Millstone 3	3411	4	Upflow
18	South Texas Project 1, 2	3800	4	Upflow
19	Sequoyah 1, 2	3411	4	Downflow
20	McGuire 1, 2 Watts Bar 1	3411	4	Upflow
21	Catawba 1, 2	3411	4	Upflow

Table 4-3
Plant Groupings for Combustion Engineering-Designed Plants

Plant Group	Plant Name	Original MWt	Bolted/Welded	Top or Bottom Mounted ICI
1	Palisades	2530	Bolted	Top
2	Fort Calhoun	1420	Bolted	Top
3	Calvert Cliffs 1, 2 Millstone 2 St. Lucie 1	2560	Welded	Top
4	ANO-2	2815	Welded	Top
5	SONGS 2, 3 St. Lucie 2 Waterford 3	2560 3390	Welded	Top
6	Palo Verde 1, 2, 3	3800	Welded	Bottom

4.2 Components and Material Lists

For the Westinghouse-designed plants, there are wider design variations among a larger number of plants (i.e., in comparison to the CE-designed plants). A conservative plant grouping based

primarily on the number of plant loops provides a three-group differentiation. Additional separation is obtained when original thermal output, baffle-barrel region flow design, and upper support plate configuration are considered (among other significant design variations). Additionally, there are a considerable number of more subtle design variations within the plant grouping scheme of Table 4-2 that would impact the general applicability of engineering inputs necessary to assess one or more of the materials aging degradation mechanisms (see Section 3 and MRP-175 [4]). Due to the large number of design permutations for Westinghouse plant internals, it was impractical to utilize the Table 4-2 grouping scheme. Table 4-4 identifies Westinghouse-designed plant internals components and the materials of construction for a combined group. It should be recognized that material and design variations must be considered when applying Table 4-4 to plant-specific cases.

Table 4-4
Components and Materials for Westinghouse-Designed Plants

Assembly	Sub-Assembly	Component	Material
Upper Internals Assembly	Control Rod Guide Tube Assemblies and Flow Downcomers	Anti-rotation studs and nuts	304 SS
		Bolts	316 SS
		C tubes	304 SS
		Enclosure pins	304 SS
		Upper guide tube enclosures	304 SS
		Flanges-intermediate	304 SS
		Flanges-intermediate	CF8
		Flanges-lower	304 SS
		Flanges-lower	CF8
		Flexureless inserts	304 SS
		Flexures	A X-750
		Guide plates/cards	304 SS
		Guide tube support pins	A X-750
		Guide tube support pins	316 SS
		Housing plates	304 SS
		Inserts	304 SS
		Lock bars	304 SS
		Sheaths	304 SS

Table 4-4 (continued)
Components and Materials for Westinghouse-Designed Plants

Assembly	Sub-Assembly	Component	Material
Upper Internals Assembly	Control Rod Guide Tube Assemblies and Flow Downcomers	Support pin cover plate	304 SS
		Support pin cover plate cap screws	316 SS
		Support pin cover plate locking caps and tie straps	304 SS
		Support pin nuts	A X-750
		Support pin nuts	316 SS
		Water flow slot ligaments	304 SS
	Mixing Devices	Mixing devices	CF8
	Upper Core Plate and Fuel Alignment Pins	Fuel alignment pins	316 SS
		Upper core plate	304 SS
	Upper Instrumentation Conduit and Supports	Bolting	316 SS
		Brackets, clamps, terminal blocks, and conduit straps	304 SS
		Conduit seal assembly–body, tubesheets	304 SS
		Conduit seal assembly–tubes	304 SS
		Conduits	304 SS
		Flange base	304 SS
		Locking caps	304 SS
		Support tubes	304 SS
	Upper Plenum	UHI flow column bases	CF8
		UHI flow columns	304 SS
	Upper Support Column Assemblies	Adapters	304 SS
		Bolts	316 SS
		Column bases	CF8
		Column bodies	304 SS

Table 4-4 (continued)
Components and Materials for Westinghouse-Designed Plants

Assembly	Sub-Assembly	Component	Material
Upper Internals Assembly	Upper Support Column Assemblies	Extension tubes	304 SS
		Flanges	304 SS
		Lock keys	304 SS 304L SS
		Nuts	304 SS
	Upper Support Plate Assembly	Bolts	316 SS
		Deep beam ribs	304 SS
		Deep beam stiffeners	304 SS
		Flange	304 SS
		Inverted top hat (ITH) flange	304 SS
		Inverted top hat (ITH) upper support plate	304 SS
		Lock keys	316 SS
		Ribs	304 SS
		Upper support plate	304 SS
		Upper support ring or skirt	304 SS
Lower Internals Assembly	Baffle and Former Assembly	Baffle bolting lock bars	304 SS
		Baffle-edge bolts	316 SS 347 SS
		Baffle plates	304 SS
		Baffle-former bolts	316 SS 347 SS
		Barrel-former bolts	316 SS 347 SS
		Former plates	304 SS
	Bottom Mounted Instrumentation (BMI) Column Assemblies	BMI column bodies	304 SS
		BMI column bolts	316 SS
		BMI column collars	304 SS

Table 4-4 (continued)
Components and Materials for Westinghouse-Designed Plants

Assembly	Sub-Assembly	Component	Material
Lower Internals Assembly	Bottom Mounted Instrumentation (BMI) Column Assemblies	BMI column cruciforms	CF8
		BMI column extension bars	304 SS
		BMI column extension tubes	304 SS
		BMI column lock caps	304L SS
		BMI column nuts	304 SS
	Core Barrel	Core barrel flange	304 SS
		Core barrel outlet nozzles	304 SS
		Upper core barrel	304 SS
		Lower core barrel	304 SS
	Diffuser Plate	Diffuser plate	304 SS
	Flux Thimbles (Tubes)	Flux thimble tube plugs	304 SS
		Flux thimbles (tubes)	316 SS
	Head Cooling Spray Nozzles	Head cooling spray nozzles	304 SS
	Irradiation Specimen Guides	Irradiation specimen guide	304 SS
		Irradiation specimen guide bolts	316 SS
		Irradiation specimen lock caps	304L SS
		Specimen plugs	304 SS
	Lower Core Plate and Fuel Alignment Pins	Fuel alignment pins	316 SS
		LCP-fuel alignment pin bolts	316 SS
		LCP-fuel alignment pin lock caps	304L SS
		Lower core plate	304 SS
		XL lower core plate	304 SS
	Lower Support Column Assemblies	Lower support column bodies	CF8
		Lower support column bodies	304 SS
		Lower support column bolts	304 SS

Table 4-4 (continued)
Components and Materials for Westinghouse-Designed Plants

Assembly	Sub-Assembly	Component	Material
Lower Internals Assembly	Lower Support Column Assemblies	Lower support column nuts	304 SS
		Lower support column sleeves	304 SS
	Lower Support Casting or Forging	Lower support casting	CF8
		Lower support forging	304 SS
	Neutron Panels/Thermal Shield	Neutron panel bolts	316 SS
		Neutron panel lock caps	304 SS
		Thermal shield bolts	316 SS
		Thermal shield dowels	316 SS
		Thermal shield flexures	304 SS
		Thermal shield or neutron panels	304 SS
	Radial Support Keys	Radial support key bolts	304 SS
		Radial support key lock keys	304 SS
		Radial support keys	304 SS
	Secondary Core Support (SCS) Assembly	SCS base plate	304 SS
		SCS bolts	316 SS
		SCS energy absorber	304 SS
		SCS guide post	304 SS
		SCS housing	304 SS
		SCS lock keys	304 SS
Interfacing Components	Interfacing Components	Clevis insert bolts	A X-750
		Clevis insert lock keys	A 600
		Clevis insert lock keys	316 SS
		Clevis inserts	A 600
		Clevis inserts	304 SS
		Clevis inserts	Stellite

Table 4-4 (continued)
Components and Materials for Westinghouse-Designed Plants

Assembly	Sub-Assembly	Component	Material
Interfacing Components	Interfacing Components	Head and vessel alignment pin bolts	316 SS
		Head and vessel alignment pin lock cups	304L SS
		Head and vessel alignment pins	304 SS
		Internals hold-down spring	304 SS
		Internals hold-down spring	403 SS
		Upper core plate alignment pins	304 SS

Six different plant groups were identified for the Combustion Engineering-designed plants, Table 4-3. These groupings were based on two primary sorting criteria–core shroud design and ICI design–further distinguished by secondary criteria of original thermal output and a corresponding similarity in component design for the balance of the reactor internals. This level of differentiation (six groups) was convenient for purposes of discussion and review during the FMECA panel evaluation process, since the panel (comprised of designers, stress analysts, thermal-hydraulics engineers, materials engineers, and neutronics analysts) could provide engineering judgments about component design, material, and operating conditions that were accurate and applicable to each group or sub-group. Table 4-5 identifies Combustion Engineering-designed plant internals and the materials of construction. It should be recognized that material and design variations must be considered when applying Table 4-5 to plant-specific cases.

Table 4-5
Components and Materials for CE-Designed Plants

Assembly	Component	Material
Upper Internals Assembly	Upper guide structure support plate	304 SS
	Upper guide structure support flange–upper	304 SS
	Upper guide structure support flange–lower	304 SS
	Cylindrical skirt	304 SS
	Grid plate	304 SS
	Control rod shroud–grid ring	304 SS
	Control rod shroud–grid beams	304 SS

Table 4-5 (continued)
Components and Materials for CE-Designed Plants

Assembly	Component	Material
Upper Internals Assembly	Control rod shroud–cross braces	304 SS
	Control rod shroud–bolts	316 SS
	GSSS guide structure plate	304 SS
	GSSS support cylinder	304 SS
	GSSS studs	316 SS
	GSSS spherical washer sets	UNS S21800
	Flange blocks	304 SS
	Flange block bolts	410 SS
	Flange block shear pins	A286 SS
	RVLMS support structure tubes	304 SS
	Fuel alignment plate	304 SS
	Fuel bundle guide pins	316 SS
	Fuel bundle guide pin nuts	304 SS
	Hold-down ring	403 SS F6NM
	Belleville washer	Alloy 718
Lower Support Structure	Core support plate	304 SS 304L SS
	Core support plate bolts	316 SS
	Core support plate dowel pins	304 SS
	Anchor block bolts	316 SS
	Anchor block dowel pins	304 SS
	Fuel alignment pins	304 SS
	Fuel alignment pins	A286 SS
	Core support columns	304 SS
	Core support columns	CF8

Table 4-5 (continued)
Components and Materials for CE-Designed Plants

Assembly	Component	Material
Lower Support Structure	Core support beams	304 SS
	Core support deep beams	304 SS
	Core support column bolts	316 SS
	Bottom plate	304 SS
	ICI support columns	304 SS
Control Element Assembly (CEA)–Shroud Assemblies	CEA shrouds	304 SS
	CEA shrouds	CPF8 CF8
	CEA shroud bases	304 SS
	CEA shroud bases	CF8
	CEA shroud extension shaft guides	304 SS
	Modified CEA shroud extension shaft guides	CF8
	Instrument tubes	304 SS
	Internal/external spanner nuts	304 SS
	CEA shroud bolts	A286 SS
	CEA shroud tie rods	304 SS
	Snubber blocks	304 SS
	Snubber shims	XM-29
	Shim bolts	316 SS
Core Support Barrel Assembly	Upper cylinder	304 SS
	Lower cylinder	304 SS
	Upper core barrel flange	304 SS
	Lower core barrel flange	304 SS
	Core barrel snubber lugs	304 SS 321 SS 348 SS
	Core barrel snubber lug bolts	316 SS

Table 4-5 (continued)
Components and Materials for CE-Designed Plants

Assembly	Component	Material
Core Support Barrel Assembly	Core barrel snubber lug bolts	A286 SS
	Alignment keys	A286 SS
	Alignment keys	304 SS
	Alignment key dowel pins	304 SS
	Core barrel outlet nozzles	304 SS
	Thermal shield	304 SS
	Thermal shield positioning pins	UNS S21800
	Thermal shield support pins	304 SS
	Shroud plates	304 SS
	Former plates	304 SS
	Ribs	304 SS
	Rings	304 SS
	Core shroud bolts	316 SS
	Barrel-core shroud bolts	316 SS
	Core shroud tie rods	348 SS
	Core shroud tie rod nuts	316 SS
	Guide lugs	304 SS 348 SS
	Guide lug insert bolts	A286 SS
	Guide lug inserts	304 SS 321 SS 348 SS
In-Core Instrumentation (ICI)	ICI guide tubes	316 SS
	ICI nozzle support plate	304 SS
	ICI thimble support plate	304 SS
	ICI thimble tubes-upper	304 SS
	ICI thimble tubes-lower	Zircaloy-4

4.3 Components Characterization and Data Collection

The reactor internals components, the material(s) of construction, and the assemblies (or subassemblies) to which they belong were identified in Tables 4-4 and 4-5. The screening process requires information on the service conditions for every component in the reactor internals; this information was generated as part of this project or it was compiled from existing sources. Three data collection efforts were undertaken:

1. Compilation from existing sources
2. Interviews with internals designers and stress analysts
3. Evaluation of neutron fluence and heat generation data

These data were used as input to the screening process, the results of which are presented in Section 5.

4.3.1 Designer/Stress Analyst Interviews

Extensive analyses have been completed that demonstrate the mechanical integrity of the reactor internals under various loading and operating conditions. The ASME Boiler and Pressure Vessel Code requires that the core support structures must survive a wide range of operating conditions and off-normal transients. Normal steady-state operating conditions generally do not challenge the integrity of the internals. Therefore, there has been no comprehensive effort to explicitly determine and record normal, steady-state operating stresses for all of the reactor internals components. Similarly, fatigue life calculations are based on conservative assumptions of the loading frequencies and have been computed for only a limited number of reactor internals locations in compliance with licensing and regulatory design requirements. A team of experienced Westinghouse stress analysts and designers was assembled to affirm the applicability of the available information for the screening evaluations.

The designer and stress analyst team was asked to consider six basic questions:

1. Could the operating stress be $\geq$30 ksi?
2. Where is the component located relative to the core?
3. Is there potential for wear?
4. Could the Cumulative Fatigue Usage Factor (CUF) be $\geq$0.1 at 40 years?
5. Does it contain a structural weld?
6. Is the component bolted or is it a spring?

The team was supplied with a list of components and a summary of results from available analyses of stress and fatigue. In the evaluation of the operating stress, the team was instructed to provide estimates of the maximum tensile stress on the surface of the component due to normal operating stresses. (The FMECA Expert Panel considered weld residual stresses and the potential

affect of transient stresses in evaluating the likelihood of failure.) Stress, for example, is a screening parameter for stress corrosion cracking, with a minimum screening value of 30 ksi for wrought austenitic stainless steel (which comprises the majority of reactor internals materials). If the team concluded that there was potential for the operating stress to be greater than 30 ksi, an affirmative answer was recorded for the first question. Additional information was collected from the team to assign stress values for components that were not fabricated from wrought austenitic stainless steel in order to recognize the fact that different criteria were established for the various types of material. In this process, it is recognized that an uncertainty arises with regard to the "total effective stress" because of the difficulty in establishing reasonable and accurate estimates of the residual stresses; this may have, for example, particular applicability to weld locations that did not experience a post-weld stress relief.

The location of the component relative to the core is important in that it influences the subsequent fluence estimation process for fluence and heat generation rates. Components that have experienced measurable wear in service, or have the possibility for contact and relative motion, were noted as having a potential for wear degradation. Fatigue usage factors from available stress reports, including analyses for power uprate, were used as the starting point for considering fatigue. The team was asked to identify the locations of all structural welds in the internals components. Lastly, the team was asked to identify all bolted joints, springs and other components for which preloads may be required to maintain reliable functionality.

From the information supplied, the team was also asked to discuss the data for each component on the list and reach consensus answers to the six questions cited above. The consensus judgments were input to a database for the subsequent screening analysis.

The operating parameters and the results of the analyst interviews are presented in Appendix A.

4.3.2 Heat Generation and Neutron Fluence

Heat generation and neutron fluence estimates require detailed results from closely coupled neutron and gamma radiation transport codes. For the screening evaluation, pre-existing analyses were used as the basis for these estimates. These analyses utilized current state of the art calculational techniques that have been benchmarked for applicability to the calculation of neutron fluence, dpa, and nuclear heat generation rates.

The analytical methodology used in the screening evaluations was based on the application of two-dimensional discrete ordinates techniques using the DORT [8] transport code and the BUGLE-96 [9] cross-section library. The DORT code and the BUGLE-96 cross-section library have been benchmarked by comparison with a large data base of PWR measurements obtained from in-vessel surveillance capsules, ex-vessel dosimetry measurements, and retrospective dosimetry obtained from samples extracted from fuel assembly components above the reactor core. The methodology has been approved by the USNRC for application to PWR surveillance capsule and pressure vessel exposure evaluations. [10]

Two typical representative plants (one Westinghouse and one CE) were selected based on the availability of detailed neutron fluence and heat generation data. The geometric models and operating histories of these two plants were chosen such that, in the context of the component screening, the resultant radiation environments would be sufficient to assure that critical components for any reactors in the Westinghouse or CE fleet would not be screened from further consideration. Recent operating experience was used to project the values to 60 years of reactor operation. Detailed fluence maps were generated for the reactor internals in the core beltline region. For the purposes of analysis, six distinct fluence regions were defined:

Region 1:		$\phi t < 1x10^{20}$ n/cm^2
Region 2:	$1x10^{20}$ n/cm^2 (0.15 dpa) $\leq$	$\phi t < 7x10^{20}$ n/cm^2
Region 3:	$7x10^{20}$ n/cm^2 (1 dpa) $\leq$	$\phi t < 1x10^{21}$ n/cm^2
Region 4:	$1x10^{21}$ n/cm^2 (1.5 dpa) $\leq$	$\phi t < 1x10^{22}$ n/cm^2
Region 5:	$1x10^{22}$ n/cm^2 (15 dpa) $\leq$	$\phi t < 5x10^{22}$ n/cm^2
Region 6:	$5x10^{22}$ n/cm^2 (75 dpa) $\leq$	ϕt

Where,

ϕt (fluence) is for neutron energies with E > 1 MeV.

The region boundaries were selected to correspond approximately with the fluence values stated in the screening criteria. The fluence regions for the core baffle/shrouds, formers, core barrel, thermal shield/neutron pad, lower core plate and upper core plate, along with directly associated parts, were assigned by inspection of fluence maps similar to the Westinghouse core baffle plate map illustrated in Figure 4-1. The upper frame illustrates the top portion of the baffle assembly (approximated as a cylinder). The lower frame illustrates the mid-core portion of the baffle assembly. Comparable illustrations for the Combustion Engineering core shroud assembly are provided in Figure 4-2. The upper frame illustrates the core side of the shroud assembly. The lower frame illustrates the outside of the shroud assembly including the former plates.

In the screening database, each component has six associated Boolean (true/false) values indicating which fluence regions intersect the component. Components above the upper core plate and below the lower core plate were defined in terms of their distance above or below, respectively, the corresponding core plate. These distances were then compared to the fluence map region boundaries to characterize the components. Radial effects also exist and will make the neutron fluence in a component near the core center higher than the fluence near the core periphery. However, the component list considers components, such as control rod guides, as a generic identification and does not specify radial locations. Therefore the screening analysis treats entire sets of components as a single item. In the subsequent phase of this program, more detailed information will be used if required to evaluate specific components or locations.

A similar strategy was used for defining heat generation rate regions, although heat generation rates (and corresponding temperature distributions) were not used quantitatively in the screening

evaluation. In the subsequent phases of this program the functionality assessments will utilize the heat generation rate information explicitly. The heat generation rate regions for the Westinghouse core baffle plate are illustrated in Figure 4-3. The upper frame illustrates the top portion of the baffle assembly (approximated as a cylinder). The lower frame illustrates the mid-core portion of the baffle assembly. Comparable illustrations for the Combustion Engineering core shroud assembly are provided in Figure 4-4. The upper frame illustrates the core side of the shroud assembly. The lower frame illustrates the outside of the shroud assembly including the former plates.

The fluence and heat generation rate evaluation was implemented to provide guidance to a generic screening evaluation for potential reactor internals degradation mechanisms. In order for the plant-specific fluence and heat generation rate values to be appropriate for the screening analysis, it is important that they be representative of the range of potential plant designs. Neutron fluxes at the core edge are established primarily by the core power density and fuel loading pattern. Beyond the core, determination of neutron fluxes is effectively a radiation shielding calculation. In this case, the neutron attenuation at the core edge is determined primarily by distance and the relative mixture of water and steel in the intervening space. The applicability of the calculated neutron flux distributions is determined by the variations in the following key parameters:

Core Power Density: The core power density that was assumed for projecting the fluence values for the Westinghouse plant was 104.5 W/cm^3. The core power density assumed in the analysis of the CE-designed plant was 83.0 W/cm^3. Higher core power densities result in higher neutron fluence values.

Core Loading Pattern: Most currently-operating plants use a low leakage core design to limit neutron fluence to the reactor vessel. Use of the low leakage core design effectively minimizes the maximum fluence to the internals surrounding the core.

Water/Steel Mixture: The composition of the effective shielding medium can vary from location to location in the internals. Some of these localized effects can be observed in the fluence maps.

Distance: Although the core size and vessel diameter clearly vary with plant capacity, there is less variation in the component locations when they are referenced to the core edge. The fuel-to-baffle gaps tend to be small and consistent from plant to plant. Baffle-to-barrel spacing will vary from design to design, but the overall effect is small. Above and below the core, the designs are remarkably consistent from plant to plant. A critical spacing that can vary from plant to plant is the distance between the active fuel and the lower core plate. This spacing is determined by the lower fuel nozzle design and will not change (except in the event of a fuel assembly re-design) significantly during the life of the plant.

All data compiled in the preceding collection steps was subsequently processed for use in the screening assessment to establish the matrix of components and their susceptibility to the materials aging degradation mechanisms, as discussed in the next section. The screening input

parameters are summarized in Tables 4-6 and 4-7 for Westinghouse and Combustion Engineering-designed plants, respectively. Additional information used in the screening is summarized in Appendix A, Tables A-1 and A-2.

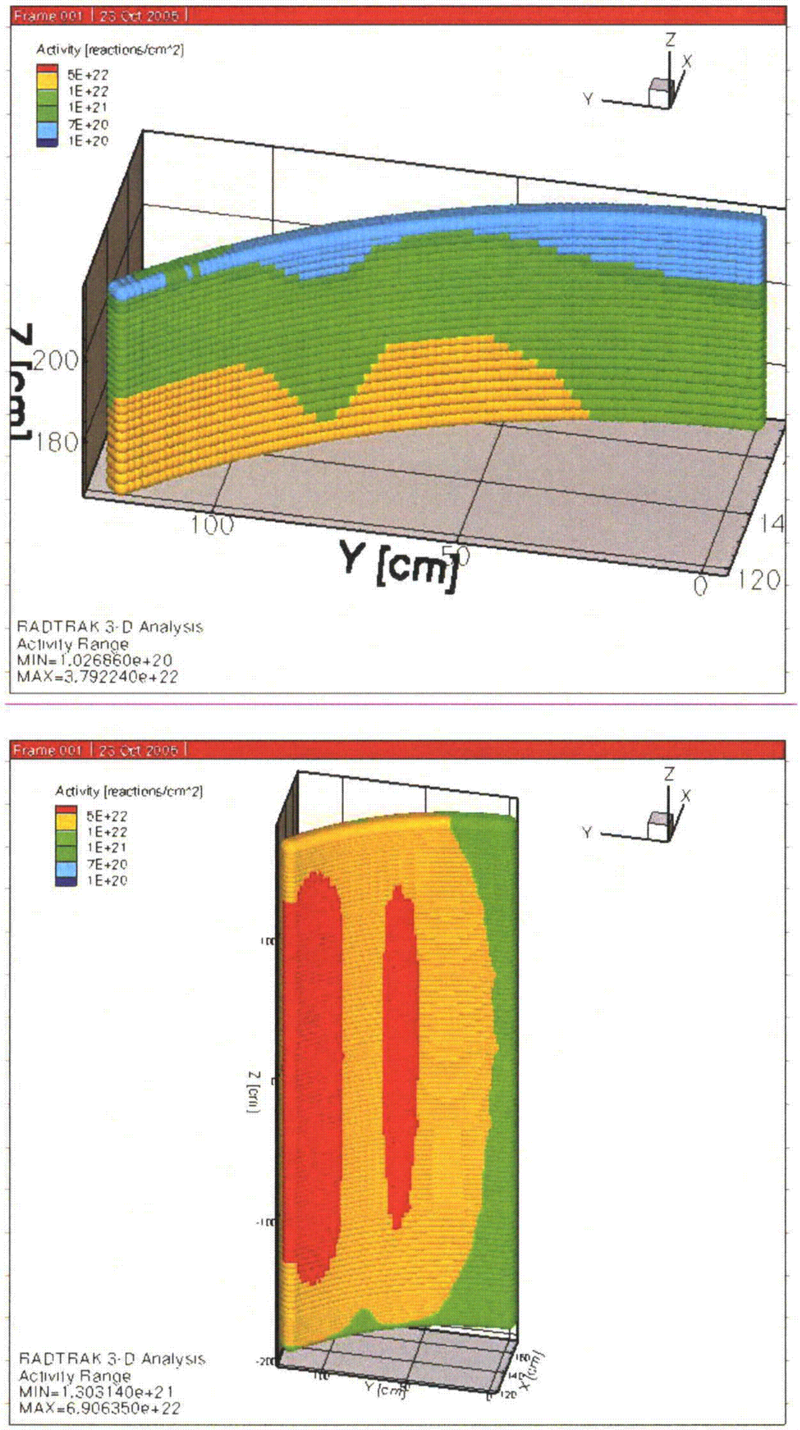
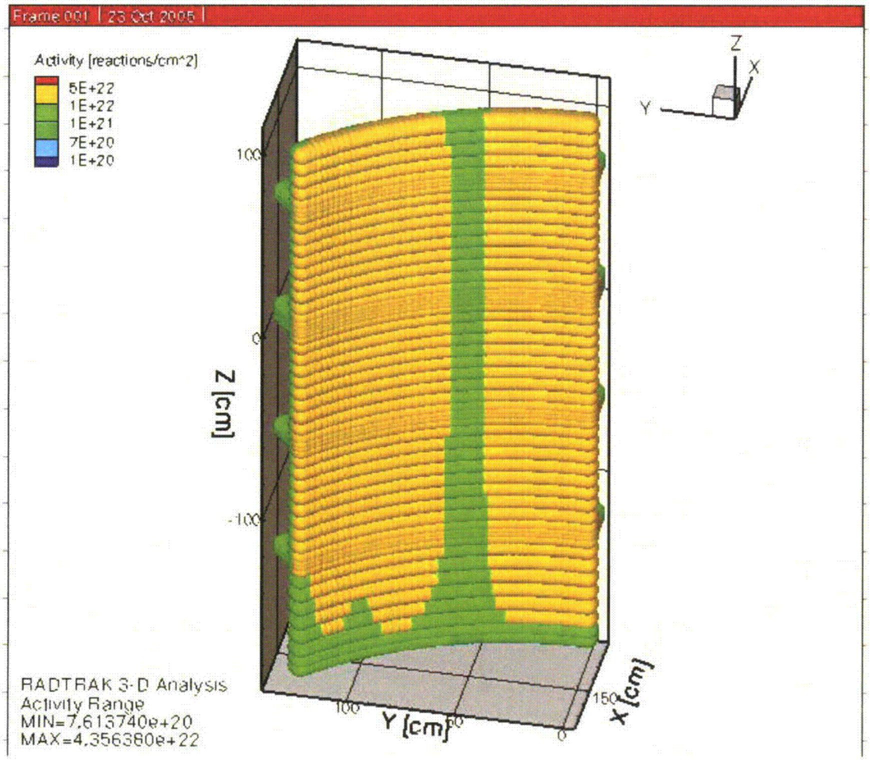


Figure 4-1
60-Year Fluence Map for Upper and Mid-core Baffle Plates in a Westinghouse-Designed Plant



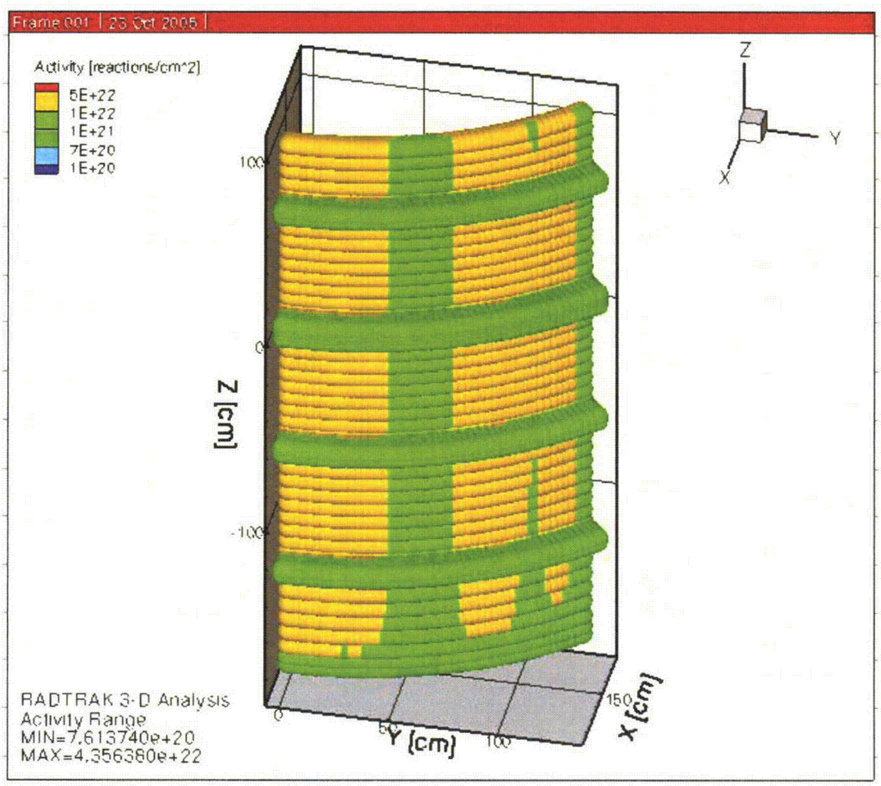
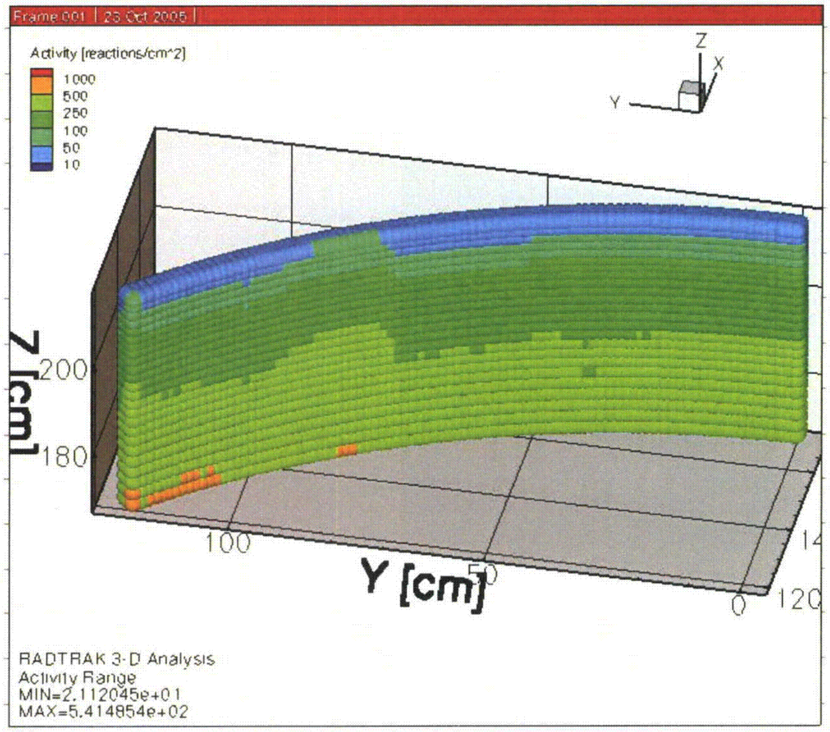


Figure 4-2
60-Year Fluence Map for Core Shroud Plates (Above) and Formers (Below) in a CE-Designed Plant



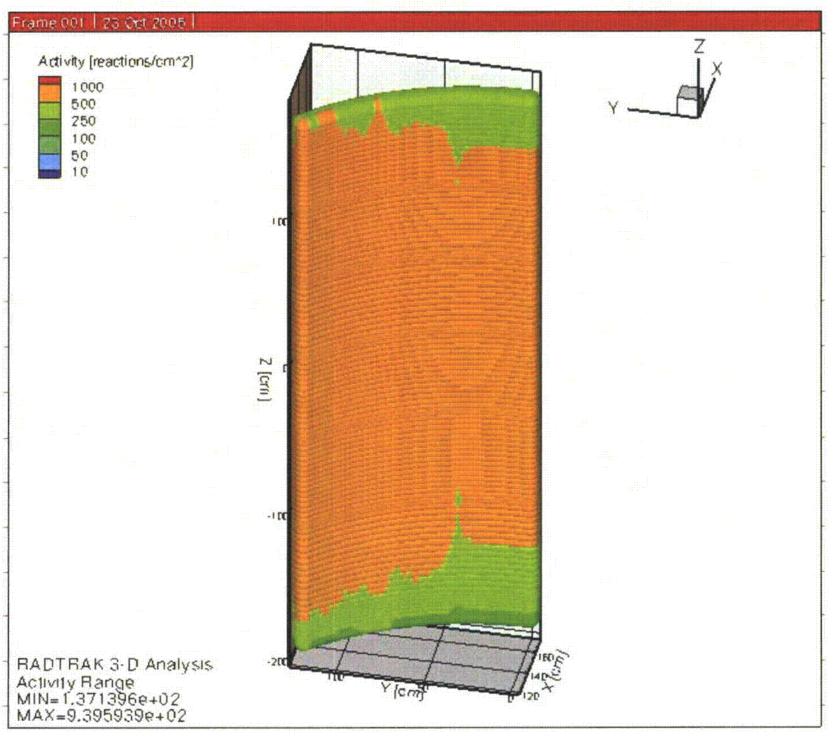


Figure 4-3
60-Year Heat Generation Map for Upper and Mid-core Baffle Plates in a Westinghouse-Designed Plant

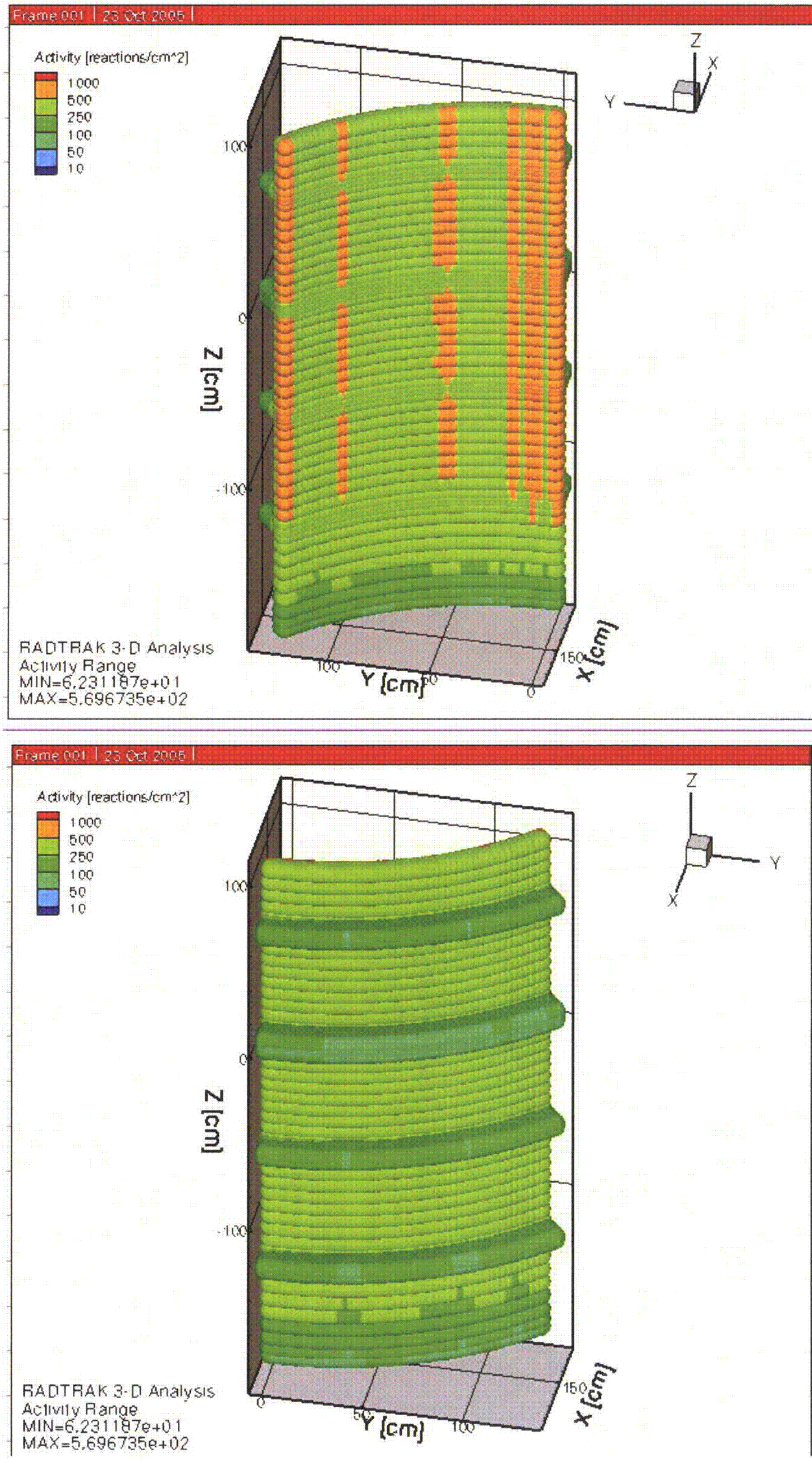


Figure 4-4
60-Year Heat Generation Map for Core Shroud Plates (Upper) and Formers (Lower) in a CE-Designed Plant

Table 4-6
Screening Input Parameters for Westinghouse-Designed Plants

Item	Material	Type or Grade/Class	Estimated[a] Temperature (°F)	Estimated Fluence Range[b] (n/cm^2, E > 1 MeV)
Upper Internals Assembly				
Control Rod Guide Tube Assemblies and Flow Downcomers				
Anti-rotation studs and nuts	Austenitic SS	304 SS	T-hot	$< 10^{20}$
Bolts	Austenitic SS	316 SS	T-hot	$< 10^{20}$
C tubes	Austenitic SS	304 SS	T-hot	7×10^{20} to 1×10^{21}
Enclosure pins	Austenitic SS	304 SS	T-hot	$< 10^{20}$
Upper guide tube enclosures	Austenitic SS	304 SS	T-hot	$< 10^{20}$
Flanges-intermediate	Austenitic SS	304 SS	T-hot	$< 10^{20}$
Flanges-intermediate	Cast Austenitic SS	CF8	T-hot	$< 10^{20}$
Flanges-lower	Austenitic SS	304 SS	T-hot	7×10^{20} to 1×10^{21}
Flanges-lower	Cast Austenitic SS	CF8	T-hot	7×10^{20} to 1×10^{21}
Flexureless inserts	Austenitic SS	304 SS	T-hot	$< 10^{20}$
Flexures	PH Ni-base Alloy	Alloy X-750	T-hot	$< 10^{20}$
Guide plates/cards	Austenitic SS	304 SS	T-hot	$< 10^{20}$
Guide tube support pins	PH Ni-base Alloy	Alloy X-750	T-hot	7×10^{20} to 1×10^{21}
Guide tube support pins	Austenitic SS	316 SS	T-hot	7×10^{20} to 1×10^{21}
Housing plates	Austenitic SS	304 SS	T-hot	$< 10^{20}$
Inserts	Austenitic SS	304 SS	T-hot	$< 10^{20}$

Table 4-6 (continued)
Screening Input Parameters for Westinghouse-Designed Plants

Item	Material	Type or Grade/Class	Estimated[a] Temperature (°F)	Estimated Fluence Range[b] (n/cm^2, E > 1 MeV)
Upper Internals Assembly				
Control Rod Guide Tube Assemblies and Flow Downcomers				
Lock bars	Austenitic SS	304 SS	T-hot	$< 10^{20}$
Sheaths	Austenitic SS	304 SS	T-hot	7×10^{20} to 1×10^{21}
Support pin cover plates	Austenitic SS	304 SS	T-hot	7×10^{20} to 1×10^{21}
Support pin cover plate cap screws	Austenitic SS	316 SS	T-hot	7×10^{20} to 1×10^{21}
Support pin cover plate locking caps and tie straps	Austenitic SS	304 SS	T-hot	7×10^{20} to 1×10^{21}
Support pin nuts	PH Ni-base Alloy	Alloy X-750	T-hot	7×10^{20} to 1×10^{21}
Support pin nuts	Austenitic SS	316 SS	T-hot	7×10^{20} to 1×10^{21}
Water flow slot ligaments	Austenitic SS	304 SS	T-hot	7×10^{20} to 1×10^{21}
Mixing Devices				
Mixing devices	Cast Austenitic SS	CF8	T-hot	7×10^{20} to 1×10^{21}
Upper Core Plate and Fuel Alignment Pins				
Fuel alignment pins	Austenitic SS	316 SS	T-hot	7×10^{20} to 1×10^{21}
Upper core plate	Austenitic SS	304 SS	T-hot	7×10^{20} to 1×10^{21}
Upper Instrumentation Conduit and Supports				
Bolting	Austenitic SS	316 SS	T-hot	$< 10^{20}$
Brackets, clamps, terminal blocks, and conduit straps	Austenitic SS	304 SS	T-hot	$< 10^{20}$

Table 4-6 (continued)
Screening Input Parameters for Westinghouse-Designed Plants

Item	Material	Type or Grade/Class	Estimated[a] Temperature (°F)	Estimated Fluence Range[b] (n/cm^2, E > 1 MeV)
Upper Instrumentation Conduit and Supports				
Conduit seal assembly–body, tubesheets	Austenitic SS	304 SS	T-hot	$< 10^{20}$
Conduit seal assembly–tubes	Austenitic SS	304 SS	T-hot	$< 10^{20}$
Conduits	Austenitic SS	304 SS	T-hot	$< 10^{20}$
Flange base	Austenitic SS	304 SS	T-hot	$< 10^{20}$
Locking caps	Austenitic SS	304 SS	T-hot	$< 10^{20}$
Support tubes	Austenitic SS	304 SS	T-hot	$< 10^{20}$
Upper Plenum				
UHI flow column bases	Cast Austenitic SS	CF8	T-hot	7×10^{20} to 1×10^{21}
UHI flow columns	Austenitic SS	304 SS	T-hot	7×10^{20} to 1×10^{21}
Upper Support Column Assemblies				
Adapters	Austenitic SS	304 SS	T-hot	$< 10^{20}$
Bolts	Austenitic SS	316 SS	T-hot	7×10^{20} to 1×10^{21}
Column bases	Cast Austenitic SS	CF8	T-hot	7×10^{20} to 1×10^{21}
Column bodies	Austenitic SS	304 SS	T-hot	7×10^{20} to 1×10^{21}
Extension tubes	Austenitic SS	304 SS	T-hot	$< 10^{20}$
Flanges	Austenitic SS	304 SS	T-hot	$< 10^{20}$
Lock keys	Austenitic SS	304 SS 304L SS	T-hot	$< 10^{20}$
Nuts	Austenitic SS	304 SS	T-hot	$< 10^{20}$

Table 4-6 (continued)
Screening Input Parameters for Westinghouse-Designed Plants

Item	Material	Type or Grade/Class	Estimated[a] Temperature (°F)	Estimated Fluence Range[b] (n/cm^2, E > 1 MeV)
Upper Support Plate Assembly				
Bolts	Austenitic SS	316 SS	T-hot	7×10^{20} to 1×10^{21}
Deep beam ribs	Austenitic SS	304 SS	T-hot	$< 10^{20}$
Deep beam stiffeners	Austenitic SS	304 SS	T-hot	$< 10^{20}$
Flange	Austenitic SS	304 SS	T-hot	$< 10^{20}$
Inverted top hat (ITH) flange	Austenitic SS	304 SS	T-hot	$< 10^{20}$
Inverted top hat (ITH) upper support plate	Austenitic SS	304 SS	T-hot	$< 10^{20}$
Lock keys	Austenitic SS	316 SS	T-hot	$< 10^{20}$
Ribs	Austenitic SS	304 SS	T-hot	$< 10^{20}$
Upper support plate	Austenitic SS	304 SS	T-hot	$< 10^{20}$
Upper support ring or skirt	Austenitic SS	304 SS	T-hot	$< 10^{20}$
Lower Internals Assembly				
Baffle and Former Assembly				
Baffle bolting lock bars	Austenitic SS	304 SS	>608	• 5×10^{22}
Baffle-edge bolts	Austenitic SS	316 SS 347 SS	>608	• 5×10^{22}
Baffle plates	Austenitic SS	304 SS	>608	• 5×10^{22}
Baffle-former bolts	Austenitic SS	316 SS 347 SS	>608	• 5×10^{22}
Barrel-former bolts	Austenitic SS	316 SS 347 SS	>608	• 5×10^{22}
Former plates	Austenitic SS	304 SS	>608	• 5×10^{22}

Table 4-6 (continued)
Screening Input Parameters for Westinghouse-Designed Plants

Item	Material	Type or Grade/Class	Estimated[a] Temperature (°F)	Estimated Fluence Range[b] (n/cm^2, E > 1 MeV)
Bottom Mounted Instrumentation (BMI) Column Assemblies				
BMI column bodies	Austenitic SS	304 SS	>608	1×10^{22} to 5×10^{22}
BMI column bolts	Austenitic SS	316 SS	T-cold	$< 10^{20}$
BMI column collars	Austenitic SS	304 SS	>608	1×10^{22} to 5×10^{22}
BMI column cruciforms	Cast Austenitic SS	CF8	>608	1×10^{22} to 5×10^{22}
BMI column extension bars	Austenitic SS	304 SS	>608	1×10^{22} to 5×10^{22}
BMI column extension tubes	Austenitic SS	304 SS	>608	1×10^{22} to 5×10^{22}
BMI column lock caps	Austenitic SS	304L SS	T-cold	$< 10^{20}$
BMI column nuts	Austenitic SS	304 SS	>608	1×10^{22} to 5×10^{22}
Core Barrel				
Core barrel flange	Austenitic SS	304 SS	T-hot	$< 10^{20}$
Core barrel outlet nozzles	Austenitic SS	304 SS	T-hot	$< 10^{20}$
Lower core barrel	Austenitic SS	304 SS	T-cold	1×10^{21} to 1×10^{22}
Upper core barrel	Austenitic SS	304 SS	T-hot	1×10^{21} to 1×10^{22}
Diffuser Plate				
Diffuser plate	Austenitic SS	304 SS	T-cold	$< 10^{20}$
Flux Thimbles (Tubes)				
Flux thimble tube plugs	Austenitic SS	304 SS	>608	1×10^{22} to 5×10^{22}
Flux thimbles (tubes)	Austenitic SS	316 SS	>608	1×10^{22} to 5×10^{22}
Head Cooling Spray Nozzles				
Head cooling spray nozzles	Austenitic SS	304 SS	T-cold	$< 10^{20}$

Table 4-6 (continued)
Screening Input Parameters for Westinghouse-Designed Plants

Item	Material	Type or Grade/Class	Estimated[a] Temperature (°F)	Estimated Fluence Range[b] (n/cm^2, E > 1 MeV)
Irradiation Specimen Guides				
Irradiation specimen guides	Austenitic SS	304 SS	T-cold	1×10^{21} to 1×10^{22}
Irradiation specimen guide bolts	Austenitic SS	316 SS	T-cold	1×10^{21} to 1×10^{22}
Irradiation specimen guide lock caps	Austenitic SS	304L SS	T-cold	1×10^{21} to 1×10^{22}
Specimen plugs	Austenitic SS	304 SS	T-cold	1×10^{21} to 1×10^{22}
Lower Core Plate and Fuel Alignment Pins				
Fuel alignment pins	Austenitic SS	316 SS	>608	1×10^{22} to 5×10^{22}
LCP-fuel alignment pin bolts	Austenitic SS	316 SS	>608	1×10^{22} to 5×10^{22}
LCP-fuel alignment pin lock caps	Austenitic SS	304L SS	>608	1×10^{22} to 5×10^{22}
Lower core plate	Austenitic SS	304 SS	>608	1×10^{22} to 5×10^{22}
XL lower core plate	Austenitic SS	304 SS	T-cold	1×10^{21} to 1×10^{22}
Lower Support Column Assemblies				
Lower support column bodies	Cast Austenitic SS	CF8	>608	1×10^{22} to 5×10^{22}
Lower support column bodies	Austenitic SS	304 SS	>608	1×10^{22} to 5×10^{22}
Lower support column bolts	Austenitic SS	304 SS	>608	1×10^{22} to 5×10^{22}
Lower support column nuts	Austenitic SS	304 SS	T-cold	$< 10^{20}$
Lower support column sleeves	Austenitic SS	304 SS	T-cold	$< 10^{20}$

Table 4-6 (continued)
Screening Input Parameters for Westinghouse-Designed Plants

Item	Material	Type or Grade/Class	Estimated[a] Temperature (°F)	Estimated Fluence Range[b] (n/cm^2, E > 1 MeV)
Lower Support Casting or Forging				
Lower support casting	Cast Austenitic SS	CF8	T-cold	< 10^{20}
Lower support forging	Austenitic SS	304 SS	T-cold	< 10^{20}
Neutron Panels/Thermal Shield				
Neutron panel bolts	Austenitic SS	316 SS	T-cold	1 x 10^{21} to 1 x 10^{22}
Neutron panel lock caps	Austenitic SS	304 SS	T-cold	1 x 10^{21} to 1 x 10^{22}
Thermal shield bolts	Austenitic SS	316 SS	T-cold	1 x 10^{21} to 1 x 10^{22}
Thermal shield dowels	Austenitic SS	316 SS	T-cold	1 x 10^{21} to 1 x 10^{22}
Thermal shield flexures	Austenitic SS	304 SS	T-cold	1 x 10^{21} to 1 x 10^{22}
Thermal shield or neutron panels	Austenitic SS	304 SS	T-cold	1 x 10^{21} to 1 x 10^{22}
Radial Support Keys				
Radial support key bolts	Austenitic SS	304 SS	T-cold	< 10^{20}
Radial support key lock keys	Austenitic SS	304 SS	T-cold	< 10^{20}
Radial support keys	Austenitic SS	304 SS	T-cold	< 10^{20}
Secondary Core Support (SCS) Assembly				
SCS base plate	Austenitic SS	304 SS	T-cold	< 10^{20}
SCS bolts	Austenitic SS	316 SS	T-cold	< 10^{20}
SCS energy absorber	Austenitic SS	304 SS	T-cold	< 10^{20}
SCS guide post	Austenitic SS	304 SS	T-cold	< 10^{20}
SCS housing	Austenitic SS	304 SS	T-cold	< 10^{20}
SCS lock keys	Austenitic SS	304 SS	T-cold	< 10^{20}

Table 4-6 (continued)
Screening Input Parameters for Westinghouse-Designed Plants

Item	Material	Type or Grade/Class	Estimated[a] Temperature (°F)	Estimated Fluence Range[b] (n/cm^2, E > 1 MeV)
Interfacing Components				
Clevis insert bolts	PH Ni-base Alloy	Alloy X-750	T-cold	$< 10^{20}$
Clevis insert lock keys	Austenitic Ni-base Alloy	Alloy 600	T-cold	$< 10^{20}$
Clevis insert lock keys	Austenitic SS	316 SS	T-cold	$< 10^{20}$
Clevis inserts	Austenitic Ni-base Alloy	Alloy 600	T-cold	$< 10^{20}$
Clevis inserts	Austenitic SS	304 SS	T-cold	$< 10^{20}$
Clevis inserts	Co Hard Facing Alloy	Stellite	T-cold	$< 10^{20}$
Head and vessel alignment pin bolts	Austenitic SS	316 SS	T-hot	$< 10^{20}$
Head and vessel alignment pin lock caps	Austenitic SS	304L SS	T-hot	$< 10^{20}$
Head and vessel alignment pins	Austenitic SS	304 SS	T-hot	$< 10^{20}$
Internals hold-down spring	Austenitic SS	304 SS	T-hot	$< 10^{20}$
Internals hold-down spring	Martensitic SS	403 SS	T-hot	$< 10^{20}$
Upper core plate alignment pins	Austenitic SS	304 SS	T-hot	7×10^{20} to 1×10^{21}

a. Temperature rise due to gamma heating was considered for components in fluence Regions 5 and 6 (1×10^{22} n/cm^2 and greater); these temperatures are indicated as > 608°F.

b. Estimate of peak fluence using the ranges stated in Section 4.2.

Table 4-7
Screening Input Parameters for Combustion Engineering-Designed Plants

Item	Material	Type or Grade/Class	Estimated[a] Temperature (°F)	Estimated Fluence Range[b] (n/cm^2, E > 1 MeV)
Upper Internals Assembly				
Upper guide structure support plate	Austenitic SS	304 SS	T-hot	$< 10^{20}$
Upper guide structure support flange-upper	Austenitic SS	304 SS	T-hot	$< 10^{20}$
Upper guide structure support flange-lower	Austenitic SS	304 SS	T-hot	$< 10^{20}$
Cylindrical skirt	Austenitic SS	304 SS	T-hot	$< 10^{20}$
Grid plate	Austenitic SS	304 SS	T-hot	$< 10^{20}$
Control rod shroud-grid ring	Austenitic SS	304 SS	T-hot	$< 10^{20}$
Control rod shroud-grid beams	Austenitic SS	304 SS	T-hot	$< 10^{20}$
Control rod shroud-cross braces	Austenitic SS	304 SS	T-hot	$< 10^{20}$
Control rod shroud-bolts	Austenitic SS	316 SS	T-hot	$< 10^{20}$
GSSS guide structure plate	Austenitic SS	304 SS	T-hot	$< 10^{20}$
GSSS support cylinder	Austenitic SS	304 SS	T-hot	$< 10^{20}$
GSSS studs	Austenitic SS	316 SS	T-hot	$< 10^{20}$
GSSS spherical washer sets	Austenitic SS	UNS S21800	T-hot	$< 10^{20}$
Flange blocks	Austenitic SS	304 SS	T-hot	$< 10^{20}$
Flange block bolts	Martensitic SS	410 SS	T-hot	$< 10^{20}$
Flange block shear pins	PH Austenitic SS	A286 SS	T-hot	$< 10^{20}$
RVLMS support structure tubes	Austenitic SS	304 SS	T-hot	1×10^{20} to 7×10^{20}

Table 4-7 (continued)
Screening Input Parameters for Combustion Engineering-Designed Plants

Item	Material	Type or Grade/Class	Estimated[a] Temperature (°F)	Estimated Fluence Range[b] (n/cm^2, E > 1 MeV)
Upper Internals Assembly				
Fuel alignment plate	Austenitic SS	304 SS	T-hot	1×10^{20} to 7×10^{20}
Fuel bundle guide pins	Austenitic SS	316 SS	T-hot	1×10^{20} to 7×10^{20}
Fuel bundle guide pin nuts	Austenitic SS	304 SS	T-hot	1×10^{20} to 7×10^{20}
Hold down ring	Martensitic SS	403 SS F6NM	T-hot	$< 10^{20}$
Belleville washer	PH Ni-base Alloy	Alloy 718	T-hot	$< 10^{20}$
Lower Support Structure				
Core support plate	Austenitic SS	304 SS 304L SS	T-cold	1×10^{21} to 1×10^{22}
Core support plate bolts	Austenitic SS	316 SS	T-cold	1×10^{21} to 1×10^{22}
Core support plate dowel pins	Austenitic SS	304 SS	T-cold	1×10^{21} to 1×10^{22}
Anchor block bolts	Austenitic SS	316 SS	T-cold	1×10^{21} to 1×10^{22}
Anchor block dowel pins	Austenitic SS	304 SS	T-cold	1×10^{21} to 1×10^{22}
Fuel alignment pins	Austenitic SS	304 SS	T-cold	1×10^{21} to 1×10^{22}
Fuel alignment pins	PH Austenitic SS	A286 SS	T-cold	1×10^{21} to 1×10^{22}
Core support columns	Austenitic SS	304 SS	T-cold	1×10^{21} to 1×10^{22}
Core support columns	Cast Austenitic SS	CF8	T-cold	1×10^{21} to 1×10^{22}
Core support beams	Austenitic SS	304 SS	T-cold	$< 10^{20}$
Core support deep beams	Austenitic SS	304 SS	T-cold	1×10^{21} to 1×10^{22}
Core support column bolts	Austenitic SS	316 SS	T-cold	1×10^{21} to 1×10^{22}
Bottom plate	Austenitic SS	304 SS	T-cold	$< 10^{20}$
ICI support columns	Austenitic SS	304 SS	T-cold	$< 10^{20}$

Table 4-7 (continued)
Screening Input Parameters for Combustion Engineering-Designed Plants

Item	Material	Type or Grade/Class	Estimated[a] Temperature (°F)	Estimated Fluence Range[b] (n/cm^2, E > 1 MeV)
Control Element Assembly (CEA) – Shroud Assemblies				
CEA shrouds	Austenitic SS	304 SS	T-hot	$< 10^{20}$
CEA shrouds	Cast Austenitic SS	CPF8 CF8	T-hot	$< 10^{20}$
CEA shroud bases	Austenitic SS	304 SS	T-hot	1×10^{20} to 7×10^{20}
CEA shroud bases	Cast Austenitic SS	CF8	T-hot	1×10^{20} to 7×10^{20}
CEA shroud extension shaft guides	Austenitic SS	304 SS	T-hot	$< 10^{20}$
Modified CEA shroud extension shaft guides	Cast Austenitic SS	CF8	T-hot	$< 10^{20}$
Instrument tubes	Austenitic SS	304 SS	T-hot	1×10^{20} to 7×10^{20}
Internal/external spanner nuts	Austenitic SS	304 SS	T-hot	$< 10^{20}$
CEA shroud bolts	PH Austenitic SS	A286 SS	T-hot	1×10^{20} to 7×10^{20}
CEA shroud tie rods	Austenitic SS	304 SS	T-hot	$< 10^{20}$
Snubber blocks	Austenitic SS	304 SS	T-hot	$< 10^{20}$
Snubber shims	Austenitic SS	XM-29	T-hot	$< 10^{20}$
Shim bolts	Austenitic SS	316 SS	T-hot	$< 10^{20}$
Core Support Barrel Assembly				
Upper cylinder	Austenitic SS	304 SS	T-hot	1×10^{20} to 7×10^{20}
Lower cylinder	Austenitic SS	304 SS	T-cold	1×10^{21} to 1×10^{22}
Upper core barrel flange	Austenitic SS	304 SS	T-hot	$< 10^{20}$
Lower core barrel flange	Austenitic SS	304 SS	T-cold	$< 10^{20}$
Core barrel snubber lugs	Austenitic SS	304 SS 321 SS 348 SS	T-cold	7×10^{20} to 1×10^{21}

Table 4-7 (continued)
Screening Input Parameters for Combustion Engineering-Designed Plants

Item	Material	Type or Grade/Class	Estimated[a] Temperature (°F)	Estimated Fluence Range[b] (n/cm^2, E > 1 MeV)
Core Support Barrel Assembly				
Core barrel snubber lug bolts	Austenitic SS	316 SS	T-cold	$< 10^{20}$
Core barrel snubber lug bolts	PH Austenitic SS	A286 SS	T-cold	$< 10^{20}$
Alignment keys	PH Austenitic SS	A286 SS	T-cold	$< 10^{20}$
Alignment keys	Austenitic SS	304 SS	T-cold	$< 10^{20}$
Alignment key dowel pins	Austenitic SS	304 SS	T-cold	$< 10^{20}$
Core barrel outlet nozzles	Austenitic SS	304 SS	T-hot	1×10^{20} to 7×10^{20}
Thermal shield	Austenitic SS	304 SS	T-cold	1×10^{20} to 7×10^{20}
Thermal shield positioning pins	Austenitic SS	UNS S21800	T-cold	1×10^{20} to 7×10^{20}
Thermal shield support pins	Austenitic SS	304 SS	T-cold	1×10^{20} to 7×10^{20}
Core Shroud Assembly				
Shroud plates	Austenitic SS	304 SS	>608	1×10^{22} to 5×10^{22}
Former plates	Austenitic SS	304 SS	>608	1×10^{22} to 5×10^{22}
Ribs	Austenitic SS	304 SS	>608	1×10^{22} to 5×10^{22}
Rings	Austenitic SS	304 SS	>608	1×10^{22} to 5×10^{22}
Core shroud bolts	Austenitic SS	316 SS	>608	1×10^{22} to 5×10^{22}
Barrel-core shroud bolts	Austenitic SS	316 SS	T-cold	1×10^{21} to 1×10^{22}
Core shroud tie rods	Austenitic SS	348 SS	T-cold	1×10^{21} to 1×10^{22}
Core shroud tie rod nuts	Austenitic SS	316 SS	T-hot	1×10^{21} to 1×10^{22}

Table 4-7 (continued)
Screening Input Parameters for Combustion Engineering-Designed Plants

Item	Material	Type or Grade/Class	Estimated[a] Temperature (°F)	Estimated Fluence Range[b] (n/cm^2, E > 1 MeV)
Core Shroud Assembly				
Guide lugs	Austenitic SS	304 SS 348 SS	T-hot	1×10^{20} to 7×10^{20}
Guide lug insert bolts	PH Austenitic SS	A286 SS	T-hot	1×10^{20} to 7×10^{20}
Guide lug inserts	Austenitic SS	304 SS 321 SS 348 SS	T-hot	1×10^{20} to 7×10^{20}
In-Core Instrumentation (ICI)				
ICI guide tubes	Austenitic SS	316 SS	T-cold	1×10^{21} to 1×10^{22}
ICI nozzle support plate	Austenitic SS	304 SS	T-cold	$< 10^{20}$
ICI thimble support plate	Austenitic SS	304 SS	T-hot	$< 10^{20}$
ICI thimble tubes-upper	Austenitic SS	304 SS	T-hot	1×10^{20} to 7×10^{20}
ICI thimble tubes-lower	Zirconium-base Alloy	Zircaloy-4	>608	• 5×10^{22}

a. Temperature rise due to gamma heating was considered for components in fluence Regions 5 and 6 (1×10^{22} n/cm^2 and greater); these temperatures are indicated as > 608°F.

b. Estimate of peak fluence using the ranges stated in Section 4.2.

5
SCREENING RESULTS FOR WESTINGHOUSE AND CE INTERNALS

5.1 Application of the Screening Criteria

The criteria established for screening the Westinghouse and Combustion Engineering internals components with respect to the age-related degradation mechanisms were presented and discussed in Section 3. The objective of this section is to describe how the screening criteria from Reference 4 were applied and to present the screening results.

The Section 3 screening criteria present a framework within which each component, complemented by considerations of specific material, stress, temperature and radiation environment, could be evaluated against the individual degradation mechanisms. In the implementation of these criteria certain simplifying assumptions were made to provide clarity. The following paragraphs present a brief summary of those simplifying assumptions. The information that follows applies solely to the screening results presented in this section.

5.1.1 Stress Corrosion Cracking

The screening criteria presented in Section 3 (Table 3-1) identified levels of effective stress, and in some cases additional metallurgical or microstructural conditions, for application to all internals components. These values vary for the different materials. In the screening process, each component was evaluated against the appropriate threshold stress for the material. However, all austenitic stainless steel components (both cast and wrought) that were judged to be heavily deformed or welded during manufacture were initially screened in regardless of the estimated effective stress. (As noted in Section 3, small weld locations such as tack welds or plug welds are excluded by MRP-175 [4,7], but each incidence identified was reviewed by the FMECA expert panel. Tables 5-1 and 5-2 report the SCC screening results due to welds following the expert panel review.) Components screened in for SCC solely on the association with a weld were differentiated from those screened in based on an estimated high level of stress. (In Tables 5-1 and 5-2, the word "weld" is included in the SCC column if the component is screened in on the basis of association with a weld.) The judgment as to the effective (tensile) stress level in the various components was elicited from Westinghouse stress analysts familiar with the design and operation of the reactor internals. Tables A-1 and A-2 indicate whether the stress for that component exceeds the screening value for the material type (see Table 3-1). All cast stainless steels identified as having stress levels above their respective criterion value (35 ksi) were screened in for SCC, without a specific appraisal of the ferrite content. Austenitic

nickel-base alloys with stress levels greater than their respective SCC screening criteria, or those components having experienced SCC degradation in service, were screened in for SCC.

5.1.2 Irradiation-Assisted Stress Corrosion Cracking (IASCC)

The IASCC screening criteria were summarized in Table 3-2. These criteria apply to all materials, and are seen to involve considerations of both stress and fluence, with the fluence expressed in terms of the damage parameter, dpa. The limited experimental data indicating conditions for which IASCC has been observed in PWR internals are summarized by the solid line in Figure 5-1. The stress and dpa values summarized in Table 3-2 were established by this curve. However, Westinghouse generated neutron fluence groupings from radiation analyses as described in Section 4.3. To be consistent with the various fluence regions defined by the radiation analyses, Westinghouse applied the simplifying assumptions shown by the dashed line in Figure 5-1. All components below 1.5 dpa were screened out for IASCC, consistent with Section 3. All components above 15 dpa were screened in for IASCC. Between 1.5 and 15 dpa, all components with effective stresses above 30 ksi were screened in for IASCC. The net effect of applying the screening criteria in this fashion is that results are slightly conservative and that end results are not impacted. All of the components that could be considered susceptible to IASCC have been identified.

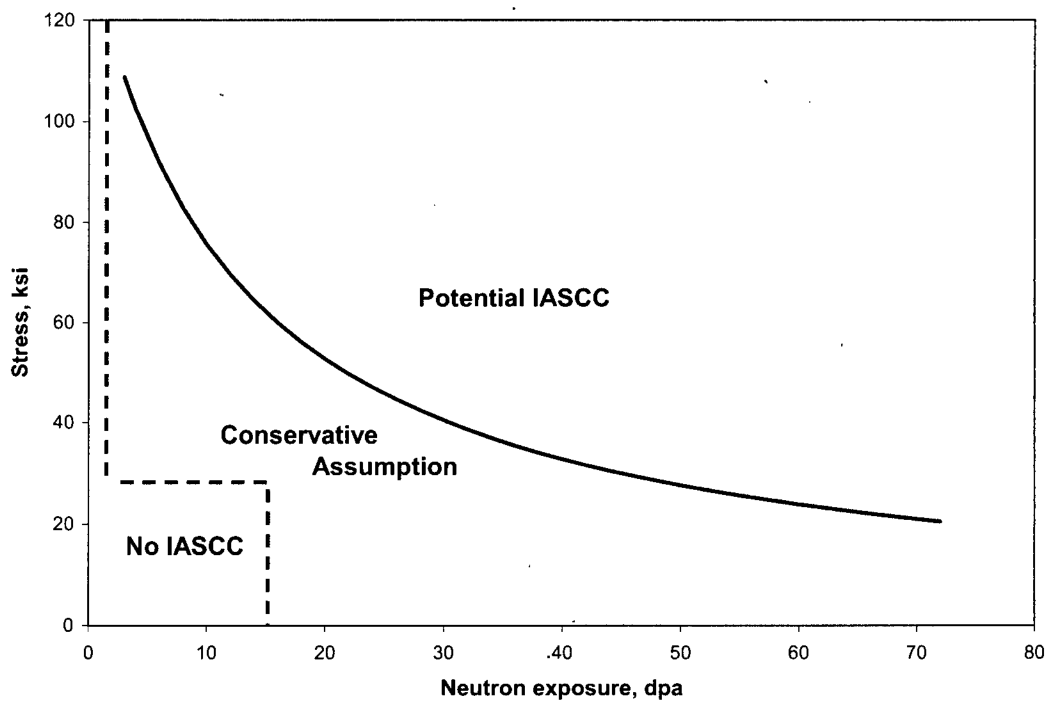


Figure 5-1
Simplifying Assumptions for IASCC Screening

5.1.3 Wear

The screening approach discussed in Section 3 was used.

5.1.4 Fatigue

The stress analysts identified components that were potentially susceptible to fatigue based on data contained in stress reports. A cumulative usage factor (CUF) of 0.1 was adopted as the screening criterion, Table 3-4.

Differentiation was made for components screened in for fatigue based purely on stress analyses from those items that were screened in for fatigue based on secondary effects arising from irradiation-induced stress relaxation.

5.1.5 Thermal Aging Embrittlement

The screening criteria for thermal aging embrittlement were summarized in Table 3-5. Westinghouse screened in all materials consistent with those criteria except that all cast austenitic stainless steels were screened in without differentiating them on the basis of delta-ferrite content. [Note that minimum and maximum delta ferrite limits in austenitic stainless steel welds and castings did not influence the screening process because all structural welds and castings were screened in for possible susceptibility to thermal aging. Actual process restrictions that may have been in place during manufacturing would have precluded the occurrence of welds or castings with delta ferrite outside the ranges for the screening criteria. For example, Regulatory Guide 1.31 on the "Control of Ferrite Content in Stainless Steel Weld Metal" (Revision 3) states that FN (ferrite number) should be between 5 and 20, whereas the screening criterion for thermal aging embrittlement is >20FN (for low molybdenum welds).]

5.1.6 Irradiation Embrittlement

The screening approach discussed in Section 3 was used.

5.1.7 Void Swelling

In Section 3 the screening criteria for neutron fluence is 1.3×10^{22} n/cm^2 [20 dpa]. A slightly more conservative screening value was used, 1.0×10^{22} n/cm^2 [15 dpa]. All components, of all material types, with fluence values over 1.0×10^{22} n/cm^2 [15 dpa] were screened in, as opposed to the limit of 1.3×10^{22} n/cm^2 [20 dpa] ascribed to austenitic stainless steel base metal and welds in MRP-175 [4]. This was done to be consistent with the various fluence regions defined by the radiation analyses, Section 4.3. It was judged that using the more conservative value (1×10^{22} n/cm^2) would have no discernible impact on the void swelling screening process. Components screened in for fluence values between 1 and 1.3×10^{22} n/cm^2 can be subsequently identified and evaluated in functionality assessments, if necessary.

Temperatures were estimated as above or below the 608°F (320°C) screening value based on consideration of gamma-heating, as reflected in Tables 4-6 and 4-7.

5.1.8 Thermal and Irradiation-induced Stress Relaxation or Irradiation Creep

The thermal stress relaxation screening approach discussed in Section 3 was used to identify all components for which "functionality requires a preload." This list of components was submitted to the FMECA Expert Panel for review. The Panel requested a technical position based on Westinghouse materials experience. A judgment was provided that supports the position that thermal stress relaxation of austenitic stainless steels at the operating temperatures of PWR reactor internals is negligible and should not be screened in for thermal stress relaxation, nor for the assumed side effects, wear and fatigue. The basis for this judgment is presented in Appendix B. This judgment does not apply to components that experience significant levels of neutron irradiation. Note that Tables 5-1 and 5-2 reflect the FMECA expert panel judgments and exclude components screened in for thermal stress relaxation.

Components that were screened in for irradiation-enhanced stress relaxation were also screened in for wear and fatigue. A positive screening for wear or fatigue on this basis indicates that further evaluation is required to judge whether there is an effect or consequence. It does not imply the component will be necessarily subject to those degradation mechanisms.

5.2 Screening Results

Complete summaries of the results of the screening process are presented in Tables 5-1 and 5-2 for Westinghouse and CE internals components, respectively. Components for which the screening process concluded there was no credible degradation mechanism are noted by an "X" in the column "None." Components that were screened in for SCC solely because of the presence of a weld are noted accordingly in the SCC column. Also, as indicated, components for which wear or fatigue are screened in solely because of concerns for irradiation effects are identified as such by a suffix (I).

Table 5-1
Screening Table for Westinghouse Reactor Internals

Assembly	Sub Assembly	Component	Material	None	SCC	IASCC	Wear	Fatigue	TE	IE	VS	ISR/IC
Upper Internals Assembly	Control Rod Guide Tube Assemblies and Flow Downcomers	Anti-rotation studs and nuts	304 SS	X								
		Bolts	316 SS	X								
		C tubes	304 SS				Wear					
		Enclosure pins	304 SS		Weld		Wear					
		Upper guide tube enclosures	304 SS		Weld							
		Flanges-intermediate	304 SS		Weld			Fat				
		Flanges-intermediate	CF8		SCC			Fat	TE			
		Flanges-lower	304 SS		Weld			Fat				
		Flanges-lower	CF8		SCC			Fat	TE	IE		
		Flexureless inserts	304 SS	X								
		Flexures	X-750		SCC							
		Guide plates/cards	304 SS		Weld		Wear	Fat				

Table 5-1 (continued)
Screening Table for Westinghouse Reactor Internals

Assembly	Sub Assembly	Component	Material	None	SCC	IASCC	Wear	Fatigue	TE	IE	VS	ISR/IC
Upper Internals Assembly	Control Rod Guide Tube Assemblies and Flow Downcomers	Guide tube support pins	X-750		SCC		Wear	Fat				I/SR
		Guide tube support pins	316 SS				Wear	Fat				I/SR
		Housing plates	304 SS	X								
		Inserts	304 SS	X								
		Lock bars	304 SS	X								
		Sheaths	304 SS				Wear					
		Support pin cover plates	304 SS	X								
		Support pin cover plate cap screws	316 SS	X								
		Support pin cover plate locking caps and tie straps	304 SS	X								
		Support pin nuts	X-750	X								

Table 5-1 (continued)
Screening Table for Westinghouse Reactor Internals

Assembly	Sub Assembly	Component	Material	None	SCC	IASCC	Wear	Fatigue	TE	IE	VS	ISR/IC
Upper Internals Assembly	Control Rod Guide Tube Assemblies and Flow Downcomers	Support pin nuts	316 SS	X								
		Water flow slot ligaments	304 SS	X								
	Mixing Devices	Mixing devices	CF8		Weld				TE	IE		
	Upper Core Plate and Fuel Alignment Pins	Fuel alignment pins	316 SS				Wear					
		Upper core plate	304 SS				Wear	Fat				
	Upper Instrumentation Conduit and Supports	Bolting	316 SS	X								
		Brackets, clamps, terminal blocks, and conduit straps	304 SS	X								
		Conduit seal assembly–body, tubesheets	304 SS	X								

Table 5-1 (continued)
Screening Table for Westinghouse Reactor Internals

Assembly	Sub Assembly	Component	Material	None	SCC	IASCC	Wear	Fatigue	TE	IE	VS	ISR/IC
Upper Internals Assembly	Upper Instrumentation Conduit and Supports	Conduit seal assembly–tubes	304 SS	X								
		Conduits	304 SS	X								
		Flange bases	304 SS	X								
		Locking caps	304 SS	X								
		Support tubes	304 SS	X								
	Upper Plenum	UHI flow column bases	CF8						TE	IE		
		UHI flow columns	304 SS	X								
	Upper Support Column Assemblies	Adapters	304 SS	X								
		Bolts	316 SS				Wear(I)	Fat(I)				I/SR
		Column bases	CF8		SCC				TE	IE		
		Column bodies	304 SS	X								

Table 5-1 (continued)
Screening Table for Westinghouse Reactor Internals

Assembly	Sub Assembly	Component	Material	None	SCC	IASCC	Wear	Fatigue	TE	IE	VS	ISR/IC
Upper Internals Assembly	Upper Support Column Assemblies	Extension tubes	304 SS		Weld							
		Flanges	304 SS	X								
		Lock keys	304 SS 304L SS	X								
		Nuts	304 SS	X								
	Upper Support Plate Assembly	Bolts	316 SS	X								
		Deep beam ribs	304 SS		Weld							
		Deep beam stiffeners	304 SS		Weld							
		Flange	304 SS	X								
		Inverted top hat (ITH) flange	304 SS		Weld			Fat				
		Inverted top hat (ITH) upper support plate	304 SS		Weld							
		Lock keys	316 SS	X								

Table 5-1 (continued)
Screening Table for Westinghouse Reactor Internals

Assembly	Sub Assembly	Component	Material	None	SCC	IASCC	Wear	Fatigue	TE	IE	VS	ISR/IC
Upper Internals Assembly	Upper Support Plate Assembly	Ribs	304 SS	X								
		Upper support plate	304 SS	X								
		Upper support ring or skirt	304 SS		Weld			Fat				
Lower Internals Assembly	Baffle and Former Assembly	Baffle bolting lock bars	304 SS			IASCC				IE	VS	
		Baffle-edge bolts	316 SS 347 SS			IASCC	Wear (I)	Fat		IE	VS	I/SR
		Baffle plates	304 SS			IASCC				IE	VS	
		Baffle-former bolts	316 SS 347 SS			IASCC	Wear (I)	Fat		IE	VS	I/SR
		Barrel-former bolts	316 SS 347 SS			IASCC	Wear (I)	Fat		IE		I/SR
		Former plates	304 SS			IASCC				IE	VS	I/SR
	Bottom Mounted Instrumentation (BMI) Column Assemblies	BMI column bodies	304 SS		Weld	IASCC		Fat		IE	VS	
		BMI column bolts	316 SS					Fat				

Table 5-1 (continued)
Screening Table for Westinghouse Reactor Internals

Assembly	Sub Assembly	Component	Material	None	SCC	IASCC	Wear	Fatigue	TE	IE	VS	ISR/IC
Lower Internals Assembly	Bottom Mounted Instrumentation (BMI) Column Assemblies	BMI column collars	304 SS			IASCC				IE	VS	
		BMI column cruciforms	CF8			IASCC			TE	IE	VS	
		BMI column extension bars	304 SS			IASCC				IE	VS	
		BMI column extension tubes	304 SS		Weld	IASCC		Fat		IE	VS	
		BMI column lock caps	304L SS	X								
		BMI column nuts	304 SS			IASCC	Wear (I)	Fat(I)		IE	VS	I/SR
	Core Barrel	Core barrel flange	304 SS		Weld							
		Core barrel outlet nozzles	304 SS		Weld			Fat				
		Lower core barrel	304 SS		Weld	IASCC				IE		

Table 5-1 (continued)
Screening Table for Westinghouse Reactor Internals

Assembly	Sub Assembly	Component	Material	None	SCC	IASCC	Wear	Fatigue	TE	IE	VS	ISR/IC
Lower Internals Assembly	Core Barrel	Upper core barrel	304 SS		Weld	IASCC				IE		
	Diffuser Plate	Diffuser plate	304 SS	X								
	Flux Thimbles (Tubes)	Flux thimble tube plugs	304 SS		Weld	IASCC				IE	VS	
		Flux thimbles (tubes)	316 SS		Weld	IASCC	Wear					
	Head Cooling Spray Nozzles	Head cooling spray nozzles	304 SS	X								
	Irradiation Specimen Guides	Irradiation specimen guides	304 SS				Wear			IE		
		Irradiation specimen guide bolts	316 SS			IASCC	Wear (I)	Fat		IE		I/SR
		Irradiation specimen guide lock caps	304L SS							IE		
		Specimen plugs	304 SS							IE		
	Lower Core Plate and Fuel Alignment Pins	Fuel alignment pins	316 SS			IASCC	Wear			IE	VS	

Table 5-1 (continued)
Screening Table for Westinghouse Reactor Internals

Assembly	Sub Assembly	Component	Material	None	SCC	IASCC	Wear	Fatigue	TE	IE	VS	ISR/IC
Lower Internals Assembly	Lower Core Plate and Fuel Alignment Pins	LCP-fuel alignment pin bolts	316 SS			IASCC	Wear	Fat(I)		IE	VS	I/SR
		LCP-fuel alignment pin lock caps	304L SS			IASCC				IE	VS	
		Lower core plate	304 SS		Weld	IASCC	Wear	Fat		IE	VS	
		XL lower core plate	304 SS		Weld	IASCC	Wear	Fat		IE		
	Lower Support Column Assemblies	Lower support column bodies	CF8			IASCC			TE	IE	VS	
		Lower support column bodies	304 SS			IASCC				IE	VS	
		Lower support column bolts	304 SS			IASCC	Wear (I)	Fat		IE	VS	I/SR

Table 5-1 (continued)
Screening Table for Westinghouse Reactor Internals

Assembly	Sub Assembly	Component	Material	None	SCC	IASCC	Wear	Fatigue	TE	IE	VS	ISR/IC
Lower Internals Assembly	Lower Support Column Assemblies	Lower support column nuts	304 SS	X								
		Lower support column sleeve	304 SS	X								
	Lower Support Casting or Forging	Lower support casting	CF8						TE			
		Lower support forging	304 SS	X								
	Neutron Panels /Thermal Shield	Neutron panel bolts	316 SS			IASCC	Wear (I)	Fat (I)		IE		I/SR
		Neutron panel lock caps	304 SS							IE		
		Thermal shield bolts	316 SS			IASCC	Wear (I)	Fat		IE		I/SR
		Thermal shield dowels	316 SS							IE		

Table 5-1 (continued)
Screening Table for Westinghouse Reactor Internals

Assembly	Sub Assembly	Component	Material	None	SCC	IASCC	Wear	Fatigue	TE	IE	VS	ISR/IC
Lower Internals Assembly	Neutron Panels /Thermal Shield	Thermal shield flexures	304 SS			IASCC	Wear (I)	Fat		IE		I/SR
		Thermal shield or neutron panels	304 SS							IE		
	Radial Support Keys	Radial support key bolts	304 SS				Wear					
		Radial support key lock keys	304 SS	X								
		Radial support keys	304 SS		Weld		Wear					
	Secondary Core Support (SCS) Assembly	SCS base plate	304 SS		Weld							
		SCS bolts	316 SS	X								
		SCS energy absorber	304 SS	X								
		SCS guide post	304 SS	X								

Table 5-1 (continued)
Screening Table for Westinghouse Reactor Internals

Assembly	Sub Assembly	Component	Material	None	SCC	IASCC	Wear	Fatigue	TE	IE	VS	ISR/IC
Lower Internals Assembly	Secondary Core Support (SCS) Assembly	SCS housing	304 SS	X								
		SCS lock keys	304 SS	X								
Interfacing Components	Interfacing Components	Clevis insert bolts	X-750		SCC		Wear					
		Clevis insert lock keys	600	X								
		Clevis insert lock keys	316 SS	X								
		Clevis inserts	600				Wear					
		Clevis inserts	304 SS				Wear					
		Clevis inserts	Stellite				Wear					
		Head and vessel alignment pin bolts	316 SS	X								
		Head and vessel alignment pin lock caps	304L SS	X								

Table 5-1 (continued)
Screening Table for Westinghouse Reactor Internals

Assembly	Sub Assembly	Component	Material	None	SCC	IASCC	Wear	Fatigue	TE	IE	VS	ISR/IC
Interfacing Components	Interfacing Components	Head and vessel alignment pins	304 SS	X								
		Internals hold-down spring	304 SS				Wear					
		Internals hold-down spring	403 SS				Wear		TE			
		Upper core plate alignment pins	304 SS		Weld		Wear					

Table 5-2
Screening Table for CE Reactor Internals

Assembly	Component	Material	None	SCC	IASCC	Wear	Fatigue	TE	IE	VS	ISR/IC
Upper Internals Assembly	Upper guide structure support plate	304 SS		Weld							
	Upper guide structure support flange-upper	304 SS		Weld		Wear					
	Upper guide structure support flange-lower	304 SS		Weld							
	Cylindrical skirt	304 SS		Weld							
	Grid plate	304 SS		Weld							
	Control rod shroud-grid ring	304 SS		Weld							
	Control rod shroud-grid beams	304 SS		Weld							
	Control rod shroud-cross braces	304 SS		Weld							
	Control rod shroud-bolts	316 SS	X								

Table 5-2 (continued)
Screening Table for CE Reactor Internals

Assembly	Component	Material	None	SCC	IASCC	Wear	Fatigue	TE	IE	VS	ISR/IC
Upper Internals Assembly	GSSS guide structure plate	304 SS		Weld							
	GSSS support cylinder	304 SS		Weld							
	GSSS studs	316 SS	X								
	GSSS spherical washer sets	UNS S21800	X								
	Flange blocks	304 SS				Wear					
	Flange block bolts	410 SS						TE			
	Flange block shear pins	A286 SS	X								
	RVLMS support structure tubes	304 SS		Weld		Wear	Fat				
	Fuel alignment plate	304 SS		Weld		Wear	Fat				
	Fuel bundle guide pins	316 SS				Wear	Fat(I)				I/SR
	Fuel bundle guide pin nuts	304 SS				Wear(I)	Fat(I)				I/SR
	Hold down ring	403 SS F6NM				Wear		TE			
	Belleville washer	718				Wear					

Table 5-2 (continued)
Screening Table for CE Reactor Internals

Assembly	Component	Material	None	SCC	IASCC	Wear	Fatigue	TE	IE	VS	ISR/IC
Lower Support Structure	Core support plate	304 SS 304L SS		Weld	IASCC	Wear	Fat		IE		
	Core support plate bolts	316 SS			IASCC	Wear(I)	Fat		IE		I/SR
	Core support plate dowel pins	304 SS							IE		
	Anchor block bolts	316 SS				Wear(I)	Fat(I)		IE		I/SR
	Anchor block dowel pins	304 SS							IE		
	Fuel alignment pins	304 SS							IE		
	Fuel alignment pins	A286 SS			IASCC	Wear(I)	Fat(I)		IE		I/SR
	Core support columns	304 SS		Weld	IASCC		Fat		IE		
	Core support columns	CF8		Weld	IASCC		Fat	TE	IE		
	Core support beams	304 SS		Weld			Fat				
	Core support deep beams	304 SS		Weld	IASCC		Fat		IE		
	Core support column bolts	316 SS			IASCC	Wear(I)	Fat(I)		IE		I/SR

**Table 5-2 (continued)
Screening Table for CE Reactor Internals**

Assembly	Component	Material	None	SCC	IASCC	Wear	Fatigue	TE	IE	VS	ISR/IC
Lower Support Structure	Bottom plate	304 SS		Weld							
	ICI support columns	304 SS		Weld							
Control Element Assembly (CEA)–Shroud Assemblies	CEA shrouds	304 SS		Weld							
	CEA shrouds	CPF8 CF8		Weld				TE			
	CEA shroud bases	304 SS		Weld							
	CEA shroud bases	CF8		Weld				TE			
	CEA shroud extension shaft guides	304 SS		Weld							
	Modified CEA shroud extension shaft guides	CF8		Weld				TE			
	Instrument tubes	304 SS		Weld			Fat				
	Internal/external spanner nuts	304 SS		Weld							
	CEA shroud bolts	A286 SS				Wear(I)	Fat(I)				I/SR
	CEA shroud tie rods	304 SS		Weld							
	Snubber blocks	304 SS		Weld							

Table 5-2 (continued)
Screening Table for CE Reactor Internals

Assembly	Component	Material	None	SCC	IASCC	Wear	Fatigue	TE	IE	VS	ISR/IC
Control Element Assembly (CEA)–Shroud Assemblies	Snubber shims	XM-29				Wear					
	Shim bolts	316 SS	X								
Core Support Barrel Assembly	Upper cylinder	304 SS		Weld							
	Lower cylinder	304 SS		Weld	IASCC				IE		
	Upper core barrel flange	304 SS		Weld		Wear					
	Lower core barrel flange	304 SS		Weld			Fat				
	Core barrel snubber lugs	304 SS 321 SS 348 SS		Weld		Wear					
	Core barrel snubber lug bolts	316 SS	X								
	Core barrel snubber lug bolts	A286 SS	X								
	Alignment keys	A286 SS				Wear					
	Alignment keys	304 SS				Wear					
	Alignment key dowel pins	304 SS	X								

Table 5-2 (continued)
Screening Table for CE Reactor Internals

Assembly	Component	Material	None	SCC	IASCC	Wear	Fatigue	TE	IE	VS	ISR/IC
Core Support Barrel Assembly	Core barrel outlet nozzles	304 SS		Weld		Wear					
	Thermal shield	304 SS		Weld							
	Thermal shield positioning pins	UNS S21800				Wear	Fat(I)				I/SR
	Thermal shield support pins	304 SS				Wear					
Core Shroud Assembly	Shroud plates	304 SS		Weld	IASCC				IE	VS	
	Former plates	304 SS		Weld	IASCC				IE	VS	
	Ribs	304 SS		Weld	IASCC				IE	VS	
	Rings	304 SS		Weld	IASCC				IE	VS	
	Core shroud bolts	316 SS			IASCC	Wear(I)	Fat		IE	VS	I/SR
	Barrel-core shroud bolts	316 SS			IASCC	Wear(I)	Fat(I)		IE		I/SR
	Core shroud tie rods	348 SS				Wear	Fat(I)		IE		I/SR
	Core shroud tie rod nuts	316 SS				Wear(I)	Fat(I)		IE		I/SR
	Guide lugs	304 SS 348 SS		Weld							

Table 5-2 (continued)
Screening Table for CE Reactor Internals

Assembly	Component	Material	None	SCC	IASCC	Wear	Fatigue	TE	IE	VS	ISR/IC
Core Shroud Assembly	Guide lug insert bolts	A286 SS				Wear(I)	Fat(I)				I/SR
	Guide lug inserts	304 SS 321 SS 348 SS				Wear					
In-Core Instrumentation (ICI)	ICI guide tubes	316 SS		Weld					IE		
	ICI nozzle support plate	304 SS		Weld							
	ICI thimble support plate	304 SS		Weld		Wear					
	ICI thimble tubes-upper	304 SS		Weld		Wear					
	ICI thimble tubes-lower	Zircaloy-4				Wear					

6 FAILURE MODES, EFFECTS AND CRITICALITY ANALYSIS

A Failure Modes, Effects Analysis (FMEA) is a bottoms-up approach to analyzing the effects on a system that may arise from the occurrence of potential failures. The following general FMEA process description is adapted from the 1987 ANSI/IEEE Standard 352 (Reference 11). The primary function of a FMEA is to consider each major part of the system, how it may fail (the mode of failure), and what the effect of the failure on the system would be (the failure effect). The objectives are to ensure that all conceivable failure modes and their effects on the operational success of the system have been considered, and to document the potential failures and evaluate the magnitude of the effects. The basic questions that are typically addressed by a FMEA include:

1. How can each part conceivably fail?
2. What mechanisms might produce these modes of failure?
3. What could the effects be if the failures occur?
4. Is the failure in the safe or unsafe direction?
5. How might the failure be detected?
6. What inherent provisions are provided in the design to compensate for the failure?

Consequences of identified potential failures may also be defined as part of the FMEA, and in some cases, a criticality ranking may be assigned. In this case the process is generally referred to as a Failure Modes, Effects, and Criticality Analysis (FMECA).

A criticality value might be based solely on the consequences or on some combination of consequences and perceived likelihood of occurrence, when sufficient information is available to make such a determination. The criticality value can then be used to rank those components that exhibit the potential to result in a system failure or might require further analysis to prevent the occurrence of modes that cause component failures.

The objective of the FMECA when coupled with the screening is to provide a technical basis to categorize and rank reactor internal components on the basis of materials degradation and functionality (including component consequence-of-failure considerations, plant reliability and financial impacts). The relationship of the FMECA to the overall functionality process was shown in Figure 2-2. It essentially provides the intermediate step between the screening and initial categorization of the reactor internal components.

A general outline of the FMECA process used for the current program for the reactor internal components is shown in Figure 6-1. The existing FMECA referenced in the upper right box of Figure 6-1 refers to WCAP-13627 prepared under the aegis of the Westinghouse Owners Group [5].

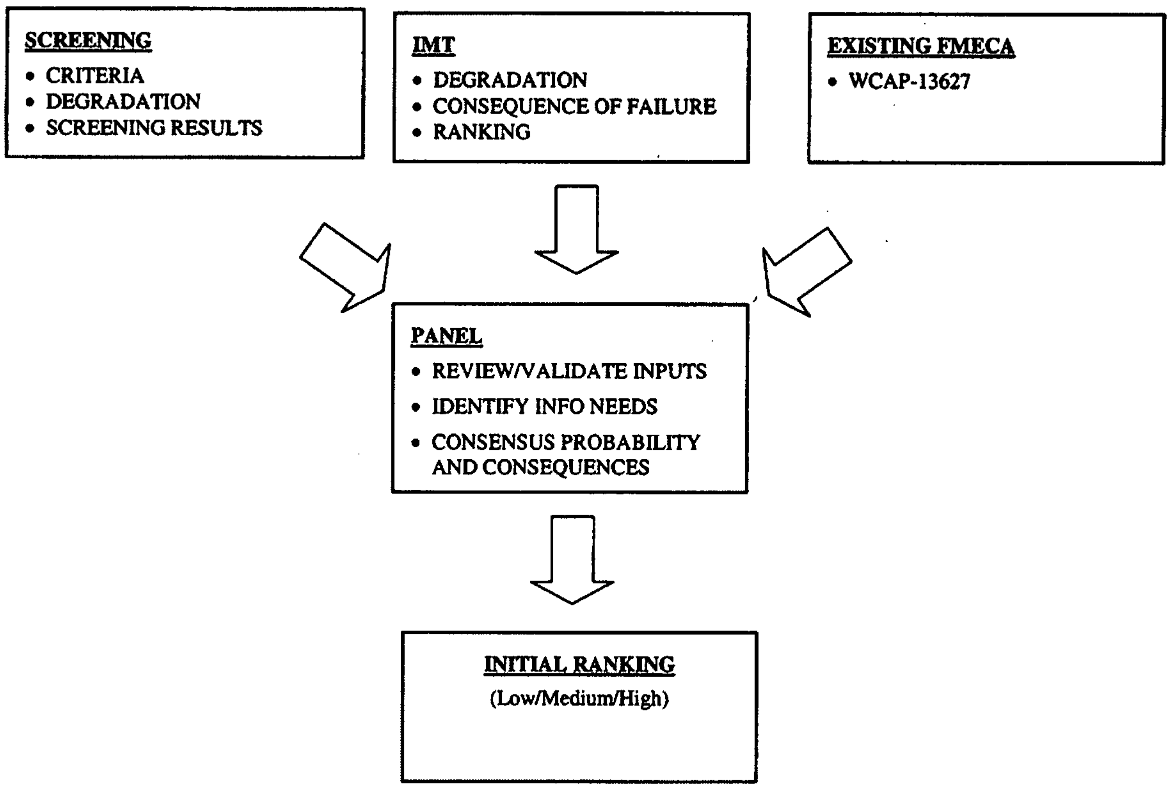


Figure 6-1
General Outline of the FMECA Process

Though differing somewhat in scope from the current program, WCAP-13627 provided valuable information to support the current effort. A description, and a first-level summary of the findings, of this previous effort are provided in Section 6.1.

6.1 Summary of Westinghouse Owners Group FMECA (WCAP-13627)

WCAP-13627 describes the methods and results of a Westinghouse Owners Group (WOG) pilot study to support the development of plant component inspection programs. The Westinghouse-designed reactor internals were selected for the pilot program. The need for inspection and prioritization of each reactor internals component was determined based on a FMECA type evaluation that estimated the likelihood of component failure and the resulting core damage and economic impact. A summary of the highest ranked internals components identified in the WOG program is provided in Table 6-1.

Table 6-1
Summary of Westinghouse Reactor Internals
Components with the Highest FMECA Ranking in WCAP-13627 [5]

Assembly (Note a)	Component Description	Core Damage[b] Rank	Economic[b,c] Rank
CS (1A)	Baffle Barrel Bolts (Down Flow)	1	1
CS (1B)	Baffle Barrel Bolts (Up Flow)	2	2
CS (2)	Baffle Formers	4	3
CS (6B)	Thermal Shield (Top Supported)	-	5
LP (14)	BMI Flux Thimble Tubes	3	6
LP (15)	BMI Support Columns	-	8
UI (17)	Fuel Pins	4	4
UI (19)	RCCA Guide Tube Support Pins	4	7
UI (20A)	RCCA Guide Tube (14X14 & 15X15)	7	-

Notes:

a. Assembly types are core support (CS), lower plenum (LP) and upper internals (UI).

b. For risk ranking, number 1 is the highest.

In reviewing the risk-based analyses presented in WCAP-13627, it is important to note that the scope of the previous effort was much more limited than that of the present program. This prior effort was intended to be a "Pilot Application" to demonstrate the feasibility of FMECA analyses for such an evaluation and considered only what were generally identifiable as major internal components whose likelihood of failure, core damage, or economic impact had been at least semiquantitatively established. For the comprehensive range of components considered in the current program, this depth of analysis did not exist, nor was it deemed to be within the purview of this program to establish or estimate similar semiquantitative assessments. As will be seen in the following discussion, assigned qualitative estimates of the likelihood such as High, Medium or Low, were adequate for the initial ranking.

6.2 Current FMECA Approach for Combustion Engineering and Westinghouse Designs

Failure modes, effects, and criticality analysis (FMECA) is a specifically designed analysis to identify the conceivable failure modes of each component and the impact of the failure on operations, the system, and surrounding components. Failure modes and causes are determined for the purpose of inspection ranking from design information, operating experience data, prior inspection results, prior risk-analysis results, and expert opinion from those with experience in operations, maintenance, inspection, and other applicable disciplines that would provide insights into the safe and reliable operation of the plant.

The FMECA begins with a qualitative analysis and is translated into a semiquantitative analysis when likelihood and consequences are estimated. Results of a FMECA yield data on each system component, failure modes, failure causes, impact on other components, and consequences. The FMECA allows systems and components to be ranked based on the likelihood and consequences. A FMECA does not serve well to identify multiple failures. Further guidance and use of FMECA for reactor vessel components is provided in MRP-134.

For the current program, the following steps were performed to complete the FMECA and initial ranking of reactor internals components. An expert panel consisting of knowledgeable Westinghouse individuals was assembled. Members with Westinghouse and CE reactor internals design experience were included as appropriate in the FMECA process. The panel provided areas of expertise in the following areas:

- Component design, testing and repair,
- Structural modeling and analysis,
- Thermal-hydraulics and systems analysis,
- Neutron fluence and radiation analysis,
- Materials degradation and failure experience,
- Component inspection experience,
- Risk assessment,
- Inspection requirements,
- System function and operating experience, and
- Licensing and regulatory interaction.

The expert panel reviewed historical data on known incidents of component malfunction and failure, and the available aging degradation data. The expert panel was provided with background and guidance on which aging mechanisms were considered significant for various reactor internals components using the screening criteria defined by the MRP.

All reactor internals components screened in for the various degradation mechanisms were identified in Section 5. Those components that were not screened in were reviewed by the experts on the panel for consensus agreement that the screening parameters were correct and had been properly applied to each component. The consensus was that the screening criteria were conservative and that some of the screened-in degradation mechanisms could have in fact been eliminated for some components.

The FMECA process was applied to both Westinghouse and Combustion Engineering reactor internals designs to initially rank the components primarily with respect to safety concerns focusing on core damage. Operability, reliability, and availability issues were also considered in the consequences.

A low consequence category was used to account for issues for those components that had a non-zero likelihood of failure to account for potential financial impacts where no specific core damage was viewed as credible.

During the current FMECA process, the responsibilities of the evaluation team were as follows:

- To review all reactor internals components for completeness, applicability and accuracy,
- To verify that component descriptions and inputs were adequate or to identify what additional information was needed to perform the evaluation,
- To verify that the likelihood and consequences identified for each component were reasonable and accurate representations,
- To provide any additional insights of which they were aware that could change either the likelihood or consequences of degradation,
- To ensure that a consistent philosophy was applied equally to all components, and
- To provide initial and final classification through a consensus process.

The review process of the expert panel members for the FMECA evaluation of each individual component in each reactor internals design included:

1. The geometry, location and function of each item were explained by the designer or analyst.
2. The material was identified and any screened-in degradation mechanisms were summarized by the materials analysts.
3. Relevant information from the previous Westinghouse Owners Group sponsored FMECA and Materials Reliability Project Issue Management Table (IMT) were considered and evaluated [1,5].
4. Any failure or degradation experience, or other known information that would affect the degradation likelihood category, was provided by knowledgeable panel members.
5. A consensus degradation category was developed by the panel, or an action item was defined to locate missing information that was needed by the panel to make their decision.

6. The effects and consequences of degradation or component failure, or other known information that would affect the damage category, was provided by knowledgeable panel members.
7. A consensus damage category was developed by the panel or an action item was defined to pursue additional information that was needed by the panel to make their decision.

For consistent application within the FMECA for reactor internals components, the following definitions and categories were used throughout:

Component Failure: Material degradation of a given component by one or more credible degradation mechanisms, as identified in the Screening Evaluation, causes the component to lose its ability to perform its intended design function either during normal operation or under accident conditions. Accident conditions include design basis earthquake or pipe break with no credit for the low likelihood of these accidents actually occurring. Cosmetic wear, craze cracking, and plastic deformation (exclusive of springs) were not considered failures.

Failure likelihood: The likelihood that component failure(s) will occur during 60 years of operation. The four categories of failure likelihood are defined in Table 6-2.

Table 6-2
Component Failure Likelihood

Category	Description	Failure History*
None	Expert panel concurs that failure of the component is not credible in a 60-year lifetime (i.e. no screened-in age-related degradation mechanisms or other evidence to support a concern).	No known failures
Low	Expert panel believes the component is unlikely to fail in a 60-year lifetime either due to known or potentially emerging issues based on current knowledge base.	No known failures
Medium	Expert panel believes there is the potential for concern, multiple degradation modes are a possibility, or believes further investigation is merited to solidify classification.	No known failures
High	Expert panel expects this component to fail or cannot exclude the possibility of failure or susceptibility to failure within the 60-year lifetime.	Known failures

Note:

* Failures were identified to be specific to the material/design combination that is recognized as having failed in service or under test conditions.

Core Damage: Physical damage to one or more fuel assemblies or other internals components-either through direct impact with the fuel, flow-jetting, loss of core support/fuel spring hold-down force, loose parts, blockage/diversion of coolant flow, or loss of insertion ability for more than one control rod that would impair the ability to safely shut down the reactor.

Damage Likelihood: The conditional likelihood that component failure(s) results in core damage given that the failure occurs irrespective of the actual failure likelihood. The four categories of conditional damage likelihood are defined in Table 6-3.

Table 6-3
Conditional Damage Likelihood

Category	Description
None	The component has no screened-in degradation mechanism. No need to assess damage probability; no financial impact.
Low	Expert panel believes there is no credible means for component failure(s) to cause damage but with potential financial impact.
Medium	Expert panel believes the potential exists for damage as a result of component (or multiple) failure(s) but that the ability to shut down the reactor in a controlled manner remains; financial impact.
High	Expert panel believes that some damage could possibly result from failure of the component(s); financial impact.

The evaluation team was charged to review the results for those components that were screened out and to concur with these judgments or to justify reinstating the component for further evaluation. The panel was charged with reviewing both the likelihood of failure of the component due to the identified degradation mechanism and the likelihood of damage as a result of the component failure, and rendering an initial assessment based on a ranking categorization of "None" (no concern, all characteristics below screening threshold values), "Low," "Medium," or "High." In all cases the panel required a 100% consensus with regard to assigning a ranking, or the component would be revisited pending review of additional inputs to provide a further basis for a decision.

The results of the FMECA were groupings of all the reactor internals components in the Westinghouse and Combustion Engineering designs into four risk groups, where each risk group was a combination of the failure and damage likelihood categories as defined in Table 6-4. The group of primary concern, prior to the Categorization and Ranking process, was Group 3 because it had the highest significance. Of secondary concern were Groups 2 (medium significance) and 1 (low significance). If the failure probability category was "None," then the Group was 0 (no significance). Reactor internals components that were screened out and put in this latter group were automatically assigned to Category A under the MRP categorization criteria (discussed in Section 7).

Table 6-4
Reactor Internals FMECA (Significance) Groups

Failure Likelihood	Consequence (Damage Likelihood)		
	Low	**Medium**	**High**
High	2	3	3
Medium	1	2	3
Low	1	1	2
None	0	0	0

The application of this logic to the full list of internals components results in the summaries presented in Tables 6-5 and 6-6 for the Westinghouse and CE designs, respectively. These tables include complete listings of all components, identify the materials and screened-in degradation mechanisms, and show the qualitative estimated probabilities of failure and damage.

For reference to the IMT document, a column listing the Consequences of Failure is also included in Tables 6-5 and 6-6 [1]. The far-right column identifies the Groups to which each component is assigned per the Table 6-4 assignment logic. (The legend or key used for the degradation mechanisms in both Tables 6-5 and 6-6 is provided following each table. Also shown is the legend for the Consequences of Failure from Appendix A of the IMT. Only those consequences indicated in Tables 6-5 and 6-6 are shown.)

Appendix A, Tables A-1 and A-2, contain additional information for each component, including the proximity of a structural weld.

The numbers of components in the non-zero significance groups for the Westinghouse and Combustion Engineering-designed internals are summarized in Tables 6-7 and 6-8, respectively.

Discussion of the results of the FMECA analysis, and the application of these results in the Categorization and Ranking process, is presented in Section 7.

Table 6-5
FMECA Results—Westinghouse Reactor Internals

Assembly	Sub-Assembly	Component	Material	IMT Conseq. of Failure	Screened-in Degradation Mechanisms	Likelihood of Failure L, M, H	Likelihood of Damage L, M, H	FMECA Group
Upper Internals Assembly	Control Rod Guide Tube Assemblies and Flow Downcomers	Anti-rotation studs and nuts	304 SS	G	NONE			0
		Bolts	316 SS	NONE	NONE			0
		C tubes	304 SS	G	3	M	M	2
		Enclosure pins	304 SS	NONE	1A, 3	L	M	1
		Upper guide tube enclosures	304 SS	NONE	1A	L	M	1
		Flanges-intermediate	304 SS	G	1A, 4	L	M	1
		Flanges-intermediate	CF8	G	1A, 4, 5	L	M	1
		Flanges-lower	304 SS	G	1A, 4	L	M	1
		Flanges-lower	CF8	G	1A, 4, 5, 6	M	M	2
		Flexureless inserts	304 SS	G	NONE			0
		Flexures	X-750	G	1	H	M	3
		Guide plates/cards	304 SS	G	1A, 3, 4	H	M	3
		Guide tube support pins	X-750	NONE	1, 3, 4, 8	H	M	3
		Guide tube support pins	316 SS	NONE	3, 4, 8	L	M	1
		Housing plates	304 SS	G	NONE			0
		Inserts	304 SS	N/A	NONE			0

Table 6-5 (continued)
FMECA Results—Westinghouse Reactor Internals

Assembly	Sub-Assembly	Component	Material	IMT Conseq. of Failure	Screened-in Degradation Mechanisms	Likelihood of Failure L, M, H	Likelihood of Damage L, M, H	FMECA Group
Upper Internals Assembly	Control Rod Guide Tube Assemblies and Flow Downcomers	Lock bars	304 SS	NONE	NONE			0
		Sheaths	304 SS	G	3	M	M	2
		Support pin cover plates	304 SS	NONE	NONE			0
		Support pin cover plate cap screws	316 SS	NONE	NONE			0
		Support pin cover plate locking caps and tie straps	304 SS	NONE	NONE			0
		Support pin nuts	X-750	NONE	NONE			0
		Support pin nuts	316 SS	NONE	NONE			0
		Water flow slot ligaments	304 SS	N/A	NONE			0
	Mixing Devices	Mixing devices	CF8	NONE	1A, 5, 6	L	L	1
	Upper Core Plate and Fuel Alignment Pins	Fuel alignment pins	316 SS	NONE	3	L	L	1
		Upper core plate	304 SS	A, G	3, 4	L	M	1
	Upper Instrumentation Conduit and Supports	Bolting	316 SS	NONE	NONE			0
		Brackets, clamps, terminal blocks, and conduit straps	304 SS	NONE	NONE			0

Table 6-5 (continued)
FMECA Results—Westinghouse Reactor Internals

Assembly	Sub-Assembly	Component	Material	IMT Conseq. of Failure	Screened-in Degradation Mechanisms	Likelihood of Failure L, M, H	Likelihood of Damage L, M, H	FMECA Group
Upper Internals Assembly	Upper Instrumentation Conduit and Supports	Conduit seal assembly–body, tubesheets	304 SS	NONE	NONE			0
		Conduit seal assembly–tubes	304 SS	NONE	NONE			0
		Conduits	304 SS	NONE	NONE			0
		Flange bases	304 SS	NONE	NONE			0
		Locking caps	304 SS	NONE	NONE			0
		Support tubes	304 SS	NONE	NONE			0
	Upper Plenum	UHI flow column bases	CF8	G	5, 6	L	L	1
		UHI flow columns	304 SS	G	NONE			0
	Upper Support Column Assemblies	Adapters	304 SS	G	NONE			0
		Bolts	316 SS	G	3, 4, 8	L	M	1
		Column bases	CF8	G	1, 5, 6	L	M	1
		Column bodies	304 SS	G	NONE			0
		Extension tubes	304 SS	G	1A	L	M	1
		Flanges	304 SS	G	NONE			0

Table 6-5 (continued)
FMECA Results—Westinghouse Reactor Internals

Assembly	Sub-Assembly	Component	Material	IMT Conseq. of Failure	Screened-in *Degradation* Mechanisms	Likelihood *of Failure* L, M, H	Likelihood *of Damage* L, M, H	*FMECA* Group
Upper Internals Assembly	Upper Support Column Assemblies	Lock keys	304 SS 304L SS	G	NONE			0
		Nuts	304 SS	G	NONE			0
	Upper Support Plate Assembly	Bolts	316 SS	NONE	NONE			0
		Deep beam ribs	304 SS	G	1A	L	M	1
		Deep beam stiffeners	304 SS	G	1A	L	M	1
		Flange	304 SS	N/A	NONE			0
		Inverted top hat (ITH) flange	304 SS	N/A	1A, 4	L	M	1
		Inverted top hat (ITH) upper support plate	304 SS	N/A	1A	L	M	1
		Lock keys	316 SS	NONE	NONE			0
		Ribs	304 SS	G	NONE			0
		Upper support plate	304 SS	G	NONE			0
		Upper support ring or skirt	304 SS	G	1A, 4	M	M	2

Table 6-5 (continued)
FMECA Results—Westinghouse Reactor Internals

Assembly	Sub-Assembly	Component	Material	IMT Conseq. of Failure	Screened-in Degradation Mechanisms	Likelihood of Failure L, M, H	Likelihood of Damage L, M, H	FMECA Group
Lower Internals Assembly	Baffle and Former Assembly	Baffle bolting lock bars	304 SS	NONE	2, 6, 7	L	L	1
		Baffle-edge bolts	316 SS 347 SS	NONE	2, 3, 4, 6, 7, 8	H	M	3
		Baffle plates	304 SS	G	2, 6, 7	M	L	1
		Baffle-former bolts	316 SS 347 SS	G	2, 3, 4, 6, 7, 8	H	L	2
		Barrel-former bolts	316 SS 347 SS	N/A	2, 3, 4, 6, 7, 8	H	L	2
		Former plates	304 SS	G	2, 6, 7	M	L	1
	Bottom Mounted Instrumentation (BMI) Column Assemblies	BMI column bodies	304 SS	G	1A, 2, 4, 6, 7	M	L	1
		BMI column bolts	316 SS	NONE	4	L	L	1
		BMI column collars	304 SS	G	2, 6, 7	M	L	1
		BMI column cruciforms	CF8	G	2, 5, 6, 7	M	L	1
		BMI column extension bars	304 SS	G	2, 6, 7	L	L	1
		BMI column extension tubes	304 SS	G	1A, 2, 4, 6, 7	M	L	1

Table 6-5 (continued)
FMECA Results—Westinghouse Reactor Internals

Assembly	Sub-Assembly	Component	Material	IMT Conseq. of Failure	Screened-in Degradation Mechanisms	Likelihood of Failure L, M, H	Likelihood of Damage L, M, H	FMECA Group
Lower Internals Assembly	Bottom Mounted Instrumentation (BMI) Column Assemblies	BMI column lock caps	304L SS	NONE	NONE			0
		BMI column nuts	304 SS	NONE	2, 3, 4, 6, 7, 8	L	L	1
	Core Barrel	Core barrel flange	304 SS	A, G	1A, 3	L	H	2
		Core barrel outlet nozzles	304 SS	G	1A, 4	M	M	2
		Lower core barrel	304 SS	A, G	1A, 2, 6	M	H	3
		Upper core barrel	304 SS	A, G	1A, 2, 6	M	H	3
	Diffuser Plate	Diffuser plate	304 SS	NONE	NONE			0
	Flux Thimbles (Tubes)	Flux thimble tube plugs	304 SS	G	1A, 2, 6, 7	M	L	1
	Head Cooling Spray Nozzles	Flux thimbles (tubes)	316 SS	G	1A, 2, 3, 6, 7	H	L	2
		Head cooling spray nozzles	304 SS	NONE	NONE			0
	Irradiation Specimen Guides	Irradiation specimen guides	304 SS	NONE	3, 6	L	L	1
		Irradiation specimen guide bolts	316 SS	NONE	2, 3, 4, 6, 8	L	L	1

Table 6-5 (continued)
FMECA Results—Westinghouse Reactor Internals

Assembly	Sub-Assembly	Component	Material	IMT Conseq. of Failure	Screened-in Degradation Mechanisms	Likelihood of Failure L, M, H	Likelihood of Damage L, M, H	FMECA Group
Lower Internals Assembly	Irradiation Specimen Guides	Irradiation specimen guide lock caps	304L SS	NONE	6	L	L	1
		Specimen plugs	304 SS	NONE	6	L	L	1
	Lower Core Plate and Fuel Alignment Pins	Fuel alignment pins	316 SS	NONE	2, 3, 6, 7	L	L	1
		LCP-fuel alignment pin bolts	316 SS	NONE	2, 3, 4, 6, 7, 8	L	L	1
		LCP-fuel alignment pin lock caps	304L SS	NONE	2, 6, 7	L	L	1
		Lower core plate	304 SS	A, F, G	1A, 2, 3, 4, 6, 7	M	M	2
		XL lower core plate	304 SS	N/A	1A, 2, 3, 4, 6	M	M	2
	Lower Support Column Assemblies	Lower support column bodies	CF8	G	2, 5, 6, 7	M	L	1
		Lower support column bodies	304 SS	G	2, 6, 7	M	L	1
		Lower support column bolts	304 SS	G	2, 3, 4, 6, 7, 8	M	L	1
		Lower support column nuts	304 SS	G	NONE			0
		Lower support column sleeves	304 SS	G	NONE			0

Table 6-5 (continued)
FMECA Results—Westinghouse Reactor Internals

Assembly	Sub-Assembly	Component	Material	IMT Conseq. of Failure	Screened-in Degradation Mechanisms	Likelihood of Failure L, M, H	Likelihood of Damage L, M, H	FMECA Group
Lower Internals Assembly	Lower Support Casting or Forging	Lower support casting	CF8	A, G	5	L	H	2
		Lower support forging	304 SS	A, G	NONE	L	H	0
	Neutron Panels/Thermal Shield	Neutron panel bolts	316 SS	NONE	2, 3, 4, 6, 8	L	L	1
		Neutron panel lock caps	304 SS	NONE	6	L	L	1
		Thermal shield bolts	316 SS	NONE	2, 3, 4, 6, 8	L	L	1
		Thermal shield dowels	316 SS	NONE	6	L	L	1
		Thermal shield flexures	304 SS	N/A	2, 3, 4, 6, 8	M	L	1
		Thermal shield or neutron panels	304 SS	G	6	L	L	1
	Radial Support Keys	Radial support key bolts	304 SS	G	3	L	L	1
		Radial support key lock keys	304 SS	G	NONE			0
		Radial support keys	304 SS	G	1A, 3	L	L	1

Table 6-5 (continued)
FMECA Results—Westinghouse Reactor Internals

Assembly	Sub-Assembly	Component	Material	IMT Conseq. of Failure	Screened-in Degradation Mechanisms	Likelihood of Failure L, M, H	Likelihood of Damage L, M, H	FMECA Group
Lower Internals Assembly	Secondary Core Support (SCS) Assembly	SCS base plate	304 SS	NONE	1A	L	L	1
		SCS bolts	316 SS	NONE	NONE			0
		SCS energy absorber	304 SS	NONE	NONE			0
		SCS guide post	304 SS	NONE	NONE			0
		SCS housing	304 SS	NONE	NONE			0
		SCS lock keys	304 SS	NONE	NONE			0
Interfacing Components	Interfacing Components	Clevis insert bolts	X-750	G	1, 3	M	L	1
		Clevis insert lock keys	Alloy 600	G	NONE			0
		Clevis insert lock keys	316 SS	G	NONE			0
		Clevis inserts	Alloy 600	G	3	L	L	1
		Clevis inserts	304 SS	G	3	L	L	1
		Clevis inserts	Stellite	G	3	L	L	1
		Head and vessel alignment pin bolts	316 SS	NONE	NONE			0
		Head and vessel alignment pin lock cups	304L SS	NONE	NONE			0

Table 6-5 (continued)
FMECA Results—Westinghouse Reactor Internals

Assembly	Sub-Assembly	Component	Material	IMT Conseq. of Failure	Screened-in Degradation Mechanisms	Likelihood of Failure L, M, H	Likelihood of Damage L, M, H	FMECA Group
Interfacing Components	Interfacing Components	Head and vessel alignment pins	304 SS	NONE	NONE			0
		Internals hold-down spring	304 SS	G	3	L	L	1
		Internals hold-down spring	403 SS	G	3, 5	L	L	1
		Upper core plate alignment pins	304 SS	NONE	1A, 3	M	L	1

Table 6-5 (continued)
FMECA Results—Westinghouse Reactor Internals

Degradation Mechanisms		Consequences of Failure*	
SCC	1	A	Precludes a safe shutdown
SCC Welds	1A	F	Breaches fuel cladding
IASCC	2	G	Causes significant economic impact
Wear	3	None	No identified consequences of failure
Fatigue	4	N/A	These are items not listed in the IMT
Thermal embrittlement	5	*More complete descriptions of these consequences are provided in the Appendix of MRP-156 [1]	
Irradiation embrittlement	6		
Void swelling	7		
Irradiation ISR/IC; Thermal SR	8		
None (Values for mechanisms did not exceed screening threshold values	None		

Table 6-6
FMECA Results—CE Reactor Internals

Assembly/ Sub-Assembly	Component	Material	IMT Conseq. of Failure	Screened-in Degradation Mechanisms	Likelihood of Failure L, M, H	Likelihood of Damage L, M, H	FMECA Group
Upper Internals Assembly	Upper guide structure support plate	304 SS	G	1A	L	M	1
	Upper guide structure support flange-upper	304 SS	G	1A, 3	L	M	1
	Upper guide structure support flange-lower	304 SS	G	1A	L	M	1
	Cylindrical skirt	304 SS	G	1A	L	M	1
	Grid plate	304 SS	G	1A	L	M	1
	Control rod shroud-grid ring	304 SS	N/A	1A	L	M	1
	Control rod shroud-grid beams	304 SS	N/A	1A	L	M	1
	Control rod shroud-cross braces	304 SS	N/A	1A	L	M	1
	Control rod shroud-bolts	316 SS	N/A	NONE			0
	GSSS guide structure plate	304 SS	N/A	1A	L	M	1
	GSSS support cylinder	304 SS	N/A	1A	L	M	1

Table 6-6 (continued)
FMECA Results—CE Reactor Internals

Assembly/ Sub-Assembly	Component	Material	IMT Conseq. of Failure	Screened-in Degradation Mechanisms	Likelihood of Failure L, M, H	Likelihood of Damage L, M, H	FMECA Group
Upper Internals Assembly	GSSS studs	316 SS	N/A	NONE			0
	GSSS spherical washer sets	UNS S21800	N/A	NONE			0
	Flange blocks	304 SS	N/A	3	L	L	1
	Flange block bolts	410 SS	N/A	5	L	L	1
	Flange block shear pins	A286 SS	N/A	NONE			0
	RVLMS support structure tubes	304 SS	N/A	1A, 3, 4	L	L	1
	Fuel alignment plate	304 SS	A, G	1A, 3, 4	M	M	2
	Fuel bundle guide pins	316 SS	N/A	3, 4, 8	L	L	1
	Fuel bundle guide pin nuts	304 SS	N/A	3, 4, 8	L	L	1
	Hold down ring	403 SS F6NM	G	3, 5	L	L	1
	Belleville washer	Alloy 718	N/A	3	L	L	1
Lower Support Structure	Core support plate	304 SS 304L SS	A, G	1A, 2, 3, 4, 6	M	M	2
	Core support plate bolts	316 SS	N/A	2, 3, 4, 6, 8	L	L	1
	Core support plate dowel pins	304 SS	N/A	6	L	L	1

Table 6-6 (continued)
FMECA Results—CE Reactor Internals

Assembly/ Sub-Assembly	Component	Material	IMT Conseq. of Failure	Screened-in Degradation Mechanisms	Likelihood of Failure L, M, H	Likelihood of Damage L, M, H	FMECA Group
Lower Support Structure	Anchor block bolts	316 SS	N/A	3, 4, 6, 8	L	L	1
	Anchor block dowel pins	304 SS	N/A	6	L	L	1
	Fuel alignment pins	304 SS	NONE	6	L	M	1
	Fuel alignment pins	A286 SS	NONE	2, 3, 4, 6, 8	M	M	2
	Core support columns	304 SS	A, G	1A, 2, 4, 6	M	L	1
	Core support columns	CF8	A, G	1A, 2, 4, 5, 6	M	L	1
	Core support beams	304 SS	A, G	1A, 4	L	L	1
	Core support deep beams	304 SS	A, G	1A, 2, 4, 6	M	M	2
	Core support column bolts	316 SS	NONE	2, 3, 4, 6, 8	M	L	1
	Bottom plate	304 SS	N/A	1A	L	L	1
	ICI support columns	304 SS	N/A	1A	L	L	1
Control Element Assembly (CEA)–Shroud Assemblies	CEA shrouds	304 SS	G	1A	L	M	1
	CEA shrouds	CPF8 CF8	G	1A, 5	L	M	1
	CEA shroud bases	304 SS	G	1A	L	M	1
	CEA shroud bases	CF8	G	1A, 5	L	M	1

Table 6-6 (continued)
FMECA Results—CE Reactor Internals

Assembly/ Sub-Assembly	Component	Material	IMT Conseq. of Failure	Screened-in Degradation Mechanisms	Likelihood of Failure L, M, H	Likelihood of Damage L, M, H	FMECA Group
Control Element Assembly (CEA)–Shroud Assemblies	CEA shroud extension shaft guides	304 SS	G	1A	L	M	1
	Modified CEA shroud extension shaft guides	CF8	G	1A, 5	L	M	1
	Instrument tubes	304 SS	NONE	1A, 4	M	L	1
	Internal/external spanner nuts	304 SS	NONE	1A	L	M	1
	CEA shroud bolts	A286 SS	NONE	3, 4, 8	L	M	1
	CEA shroud tie rods	304 SS	N/A	1A	L	M	1
	Snubber blocks	304 SS	N/A	1A	L	L	1
	Snubber shims	XM-29	N/A	3	L	L	1
	Shim bolts	316 SS	N/A	NONE			0
Core Support Barrel Assembly	Upper cylinder	304 SS	A, G	1A	L	H	2
	Lower cylinder	304 SS	A, G	1A, 2, 6	M	H	3
	Upper core barrel flange	304 SS	A, G	1A, 3	L	H	2
	Lower core barrel flange	304 SS	A, G	1A, 4	L	H	2
	Core barrel snubber lugs	304 SS 321 SS 348 SS	G	1A, 3	L	L	1

Table 6-6 (continued)
FMECA Results—CE Reactor Internals

Assembly/ Sub-Assembly	Component	Material	IMT Conseq. of Failure	Screened-in Degradation Mechanisms	Likelihood of Failure L, M, H	Likelihood of Damage L, M, H	FMECA Group
Core Support Barrel Assembly	Core barrel snubber lug bolts	316 SS	N/A	NONE			0
	Core barrel snubber lug bolts	A286 SS	N/A	NONE			0
	Alignment keys	A286 SS	NONE	3	L	L	1
	Alignment keys	304 SS	NONE	3	L	L	1
	Alignment key dowel pins	304 SS	NONE	NONE			0
	Core barrel outlet nozzles	304 SS	G	1A, 3	L	M	1
	Thermal shield	304 SS	G	1A	L	L	1
	Thermal shield positioning pins	UNS S21800	NONE	3, 4, 8	M	L	1
	Thermal shield support pins	304 SS	NONE	3	L	L	1
Core Shroud Assembly	Shroud plates	304 SS	G	1A, 2, 6, 7	M	M	2
	Former plates	304 SS	G	1A, 2, 6, 7	M	L	1
	Ribs	304 SS	G	1A, 2, 6, 7	M	L	1
	Rings	304 SS	G	1A, 2, 6, 7	M	L	1
	Core shroud bolts	316 SS	G	2, 3, 4, 6, 7, 8	M	L	1

Table 6-6 (continued)
FMECA Results—CE Reactor Internals

Assembly/ Sub-Assembly	Component	Material	IMT Conseq. of Failure	Screened-in Degradation Mechanisms	Likelihood of Failure L, M, H	Likelihood of Damage L, M, H	FMECA Group
Core Shroud Assembly	Barrel-core shroud bolts	316 SS	G	2, 3, 4, 6, 8	M	L	1
	Core shroud tie rods	348 SS	N/A	3, 4, 6, 8	M	L	1
	Core shroud tie rod nuts	316 SS	N/A	3, 4, 6, 8	M	L	1
	Guide lugs	304 SS 348 SS	NONE	1A	L	L	1
	Guide lug insert bolts	A286 SS	N/A	3, 4, 8	M	L	1
	Guide lug inserts	304 SS 321 SS 348 SS	NONE	3	L	L	1
In-Core Instrumentation (ICI)	ICI guide tubes	316 SS	NONE	1A, 6	L	L	1
	ICI nozzle support plate	304 SS	G	1A	L	L	1
	ICI thimble support plate	304 SS	G	1A, 3	L	L	1
	ICI thimble tubes-upper	304 SS	NONE	1A, 3	L	L	1
	ICI thimble tubes-lower	Zircaloy-4	NONE	3	M	L	1

Table 6-6 (continued)
FMECA Results—CE Reactor Internals

Degradation Mechanisms		Consequences of Failure *	
SCC	1	A	Precludes a safe shutdown
SCC Welds	1A	F	Breaches fuel cladding
IASCC	2	G	Causes significant economic impact.
Wear	3	None	No identified consequences of failure
Fatigue	4	N/A	These are items not listed in the IMT
Thermal embrittlement	5	*-More complete descriptions of these consequences are provided in the Appendix of MRP-156 [1]	
Irradiation embrittlement	6		
Void swelling	7		
Irradiation ISR/IC	8		
None (Values for mechanisms did not exceed screening threshold values)	None		

Table 6-7
Number of Components in the Non-zero Significance Groups*
for the Westinghouse Reactor Internals Designs

		Damage Likelihood		
		Low	**Medium**	**High**
Failure Likelihood	**High**	3	4	0
	Medium	14	7	2
	Low	25	14	2
*The higher the significance group, the darker the shading.				

Table 6-8
Number of Components in the Non-zero Significance Groups*
for the Combustion Engineering Reactor Internals Designs

		Damage Likelihood		
		Low	**Medium**	**High**
Failure Likelihood	**High**	0	0	0
	Medium	13	5	1
	Low	28	21	3
*The higher the significance group, the darker the shading.				

7 CATEGORIZATION AND RANKING

7.1 Background

As described in Section 2, the Categorization and Ranking of the internals components represents the final step that provides the basis for the activities associated with the subsequent functionality assessments. The objective of the Categorization and Ranking task was to process the results of the expert panel semiquantitative determination represented in the FMECA tabulations. A preliminary rating (i.e., assignment to Group 1, 2 or 3) based on the FMECA results was assigned to each identified component. To facilitate the categorization process, a hierarchy of risk categories was established following MRP-134 [6]. These categories are:

7.1.1 Category A

Those component items for which aging effects are below the screening criteria. Aging degradation significance is minimal.

The initial set of Category A components consisted of items for which all degradation mechanisms were initially screened out. A review by the FMECA panel endorsed this initial screening. These components are identified as "None" in the appropriate columns in Tables 6-5 and 6-6.

In addition, the FMECA results identified additional components for which age-related degradation mechanisms have minimal likelihood to cause failure. Thus, these components are also assigned to Category A. This action essentially screens these components out of further consideration as the project proceeds into the functionality assessments.

Additional components may ultimately be categorized as Category A as discussed in the Category B definition.

7.1.2 Category C

Those "lead" component items for which aging effects are above screening levels. Aging degradation significance is high or moderate. Enhanced/augmented inspections (e.g., enhanced VT-1, UT, etc.) and/or surveillance sampling typically may be warranted to assess aging effects and verify component item functionality.

All non-Category A components were initially "screened in" as Category C for further consideration. These components, for which aging effects are above the threshold values of the screening criteria, are assessed to have moderate to high likelihood of occurrence, and have the potential for significant damage. Moreover, they have not been demonstrated, analytically or by experiment, to be sufficiently damage-tolerant to remain functional relative to the aging degradation mechanism(s) identified.

Components initially placed in Category C may be assigned to Category B', as described in the subsequent discussion in this section. This separation process may be the result of expert solicitation, or may require a detailed analysis (depending on the complexity of the situation).

Those Category C components remaining were recommended for consideration in the subsequent functionality assessments.

7.1.3 Category B

Category B items are defined as those component items that also are above screening levels but are not "lead" component items. Aging degradation significance is moderate. Category B component items may require additional evaluations to be shown tolerant of the aging effects with no loss of functionality (i.e., damage tolerant).

Non-Category A components that are judged to have moderate susceptibility and potentially significant consequences, such that the effects on function cannot easily be dispositioned by screening, and yet are not considered Category C components, are assigned to Category B. Some of the components included in the Category B list may have been screened in for susceptibility to one or more degradation mechanisms, but the likelihood of occurrence and the implied safety risk were assessed by qualitative expert assessment to be low to moderate.

If it is further concluded that the existing 10-year in-service inspection or other in-place aging management plans are sufficient to preclude a safety, reliability, or financial concern, such components can be reassigned as Category A.

7.1.4 Category B'

Category B' items are defined as those "lead" component items that can be shown to be tolerant of the aging effects through a functionality assessment. These are candidate component items for an expanded inspection program. These items will not be fully identified until subsequent stages of the program are completed. For the current stage of the functionality analysis project, no Category B' components are identified.

7.2 Categorization and Ranking

The initial inputs to the categorization and ranking process were the results of the FMECA evaluation presented in Section 6. The next step was to perform a more precise ranking of all the Group 2 and 3 components again referring to the MRP-134 definitions [6]. The ranking is based on a combination of the susceptibility to aging and the potential for loss of function because of materials degradation. This process is exemplified in the following MRP-134 equation:

Relative Significance ≈ [Likelihood of Occurrence] x [Consequence of Damage]

Where:

- Likelihood of Occurrence is defined as the potential for aging degradation based on susceptibility factors (stress, fluence, temperature, material)
- Consequence of Damage is defined as the potential for loss of function given material degradation.

This produces a more focused ranking based on all of the input information. Each non-Category A component was placed into one of the three risk groups. The full list of FMECA results are presented in Tables 6-5 and 6-6 for the Westinghouse and CE designs, respectively. The far-right column of these tables identifies the initial Group to which each component is assigned per the FMECA. The numbers of components in the non-zero risk groups for the Westinghouse and Combustion Engineering-designed internals are summarized in Tables 6-7 and 6-8, respectively.

Group 3 contains those components determined to have the highest combination of likelihood of occurrence and consequence of failure–i.e., risk-damage. Groups 1 and 2 represent components determined to have relatively lower probabilities and/or consequences (Group 1 is the lowest). A total of twenty-seven components were identified as either Group 2 or Group 3 as a result of the FMECA.

A ranking of these twenty-seven components was performed by an expert panel. Each degradation mechanism was separately evaluated for significance relative to consequences, and outcomes were summed to arrive at a hybrid "score". This ranking process was implemented by the combined consideration of the following factors:

- The extent to which failure might occur due to the degradation mechanism(s) identified
- The consequences of such degradation with respect to both safety, reliability and economic risk

The components are listed in order of significance, with the most significant as item 1, along with the various degradation mechanisms identified for each component in Table 7-1. The final entry in the table is for the Combustion Engineering ICI thimble tubes. Although these tubes were assigned to Group 1 in the FMECA analysis, recent in-service degradation due to irradiation-induced growth was identified and it was deemed appropriate to include them for subsequent evaluation. These were assigned to FMECA Group 1 primarily because the issue of

irradiation-induced growth of zirconium-based materials was not recognized in MRP-175 as a degradation mechanism; accordingly, no screening criteria were provided. Thus twenty-eight components are identified in Table 7-1.

The components listed in Table 7-1 were considered as candidates for "lead" or Category C components for Phase 2. Applying the definition for Category C components stated earlier, a total of nineteen components were identified. These include the sixteen highest-ranked components, along with several additional components identified as Category C due to a consistently demonstrated history of in-service failures. This was assessed by the expert panel to lead to increased concerns for plant reliability and financial impacts. Similar application of the definitions stated earlier resulted in identifying eight of the remaining nine components as Category B. The final component, the Westinghouse lower support casting, had been identified as a FMECA Group 2 component based on the consequences of an assumed failure. However, consistent with the MRP-134 definitions, this component was placed into Category A after consideration of the very low probability of degradation and consequence due to the identified thermal embrittlement degradation mechanism.

Group 1 components, which were not explicitly evaluated in the aforementioned ranking process, were categorized based solely on the results of the FMECA evaluation since no subsequent concern has been identified. Category A and Category B assignments of these Group 1 components were dictated by their relative probability of failure as a result of degradation from all active (screened-in) age-related degradation mechanisms. Components deemed to have a moderate probability of failure were placed in Category B, while the remaining components (those with a low probability of failure) were placed in Category A. One exception is the internals hold down spring fabricated from 304 SS. Thermal "ratcheting", leading to permanent deformation, is not one of the explicitly characterized degradation mechanisms from MRP-175 but may occur in this component and reduce the spring hold-down force over time. This particular phenomenon was assessed to have a moderate likelihood of occurrence; hence, it was assigned to Category B to warrant attention during the development of Inspection and Evaluation (I&E) guidelines.

The result of this initial categorization of all Westinghouse and Combustion Engineering internals components are presented in Tables 7-2 and 7-3, respectively. For ease of reference, the FMECA Groups are also identified in these tables. It is to be emphasized that many of the initial categorizations provided in Tables 7-2 and 7-3 may be reassessed once other factors, such as the availability of mitigation programs are factored into the assessment. Figures 7-1 and 7-2 are provided to indicate the locations of each of the Category C components.

Table 7-1
Preliminary Ranking of Components in FMECA Groups 2 and 3

Rank	Design	Component	Identified Degradation Mechanisms							
			SCC	IASCC	Wear	Fatigue	TE	IE	VS	ISR/IC
1	Westinghouse	Baffle-former bolts	-	X	X	X	-	X	X	X
2	Westinghouse	Baffle-edge bolts	-	X	X	X	-	X	X	X
3	Westinghouse	Barrel-former bolts	-	X	X	X	-	X	X	X
4	CE	Shroud plates	X	X	-	-	-	X	X	-
5	Westinghouse	Lower core plate	X	X	X	X	-	X	X	-
6	Westinghouse	XL lower core plate	X	X	X	X	-	X	-	-
7	CE	Core support plate	X	X	X	X	-	X	-	-
8	Westinghouse	Guide plates/cards	X	-	X	X	-	-	-	-
9	CE	Core support deep beams	X	X	-	X	-	X	-	-
10	CE	Fuel alignment pins (A286SS)	-	X	X	X	-	X	-	X
11	Westinghouse	Guide tube support pins (Alloy X-750)	X	-	X	X	-	-	-	X
12	Westinghouse	Lower core barrel	X	X	-	-	-	X	-	-
13	Westinghouse	Upper core barrel	X	X	-	-	-	X	-	-
14	CE	Lower cylinder	X	X	-	-	-	X	-	-
15	Westinghouse	C tubes	-	-	X	-	-	-	-	-

Table 7-1 (continued)
Preliminary Ranking of Components in FMECA Groups 2 and 3

Rank	Design	Component	Identified Degradation Mechanisms							
			SCC	IASCC	Wear	Fatigue	TE	IE	VS	ISR/IC
16	Westinghouse	Sheaths	-	-	X	-	-	-	-	-
17	Westinghouse	Core barrel outlet nozzles	X	-	-	X	-	-	-	-
18	Westinghouse	Upper support ring or skirt	X	-	-	X	-	-	-	-
19	CE	Fuel alignment plate	X	-	X	X	-	-	-	-
20	CE	Lower core barrel flange	X	-	-	X	-	-	-	-
21	Westinghouse	Flanges–lower (CF8)	X	-	-	X	X	X	-	-
22	Westinghouse	Flux thimbles (tubes)	X	X	X	-	-	X	X	-
23	Westinghouse	Core barrel flange	X	-	X	-	-	-	-	-
24	CE	Upper core barrel flange	X	-	X	-	-	-	-	-
25	CE	Upper cylinder	X	-	-	-	-	-	-	-
26	Westinghouse	Flexures	X	-	-	-	-	-	-	-
27	Westinghouse	Lower support casting	-	-	-	-	X	-	-	-
-	CE	ICI thimble tubes-lower*	-	-	X	-	-	-	-	-

* This component was FMECA Group 1, but was included here because of observed field failures due to irradiation-induced growth.

Table 7-2
Categorization of Westinghouse Reactor Internals Components

Assembly	Subassembly	Component	Material	FMECA Group	Category
Upper Internals Assembly	Control Rod Guide Tube Assemblies and Flow Downcomers	Anti-rotation studs and nuts	304 SS	0	A
		Bolts	316 SS	0	A
		C Tubes	304 SS	2	C
		Enclosure pins	304 SS	1	A
		Upper guide tube enclosures	304 SS	1	A
		Flanges–intermediate	304 SS	1	A
		Flanges–intermediate	CF8	1	A
		Flanges–lower	304 SS	1	A
		Flanges–lower	CF8	2	B
		Flexureless inserts	304 SS	0	A
		Flexures	Alloy X-750	3	C
		Guide plates/cards	304 SS	3	C
		Guide tube support pins	Alloy X-750	3	C
		Guide tube support pins	316 SS	1	A
		Housing plates	304 SS	0	A
		Inserts	304 SS	0	A
		Locking bars	304 SS	0	A

Table 7-2 (continued)
Categorization of Westinghouse Reactor Internals Components

Assembly	Subassembly	Component	Material	FMECA Group	Category
Upper Internals Assembly	Control Rod Guide Tube Assemblies and Flow Downcomers	Sheaths	304 SS	2	C
		Support pin cover plates	304 SS	0	A
		Support pin cover plate caps screws	316 SS	0	A
		Support pin cover plate locking caps and tie straps	304 SS	0	A
		Support pin nuts	Alloy X-750	0	A
		Support pin nuts	316 SS	0	A
		Water flow slot ligaments	304 SS	0	A
	Mixing Devices	Mixing Devices	CF8	1	A
	Upper Core Plate and Fuel Alignment Pins	Fuel alignment pins	316 SS	1	A
		Upper core plate	304 SS	1	A
	Upper Instrumentation Conduit and Supports	Bolting	316 SS	0	A
		Brackets, clamps, terminal blocks, and conduit straps	304 SS	0	A
		Conduit seal assembly–body, tubesheets	304 SS	0	A
		Conduit seal assembly–tubes	304 SS	0	A
		Conduits	304 SS	0	A

Table 7-2 (continued)
Categorization of Westinghouse Reactor Internals Components

Assembly	Subassembly	Component	Material	FMECA Group	Category
Upper Internals Assembly	Upper Instrumentation Conduit and Supports	Flange bases	304 SS	0	A
		Locking caps	304 SS	0	A
		Support tubes	304 SS	0	A
	Upper Plenum	UHI flow column bases	CF8	1	A
		UHI flow columns	304 SS	0	A
	Upper Support Column Assemblies	Adapters	304 SS	0	A
		Bolts	316 SS	1	A
		Column bases	CF8	1	A
		Column bodies	304 SS	0	A
		Extension tubes	304 SS	1	A
		Flanges	304 SS	0	A
		Lock keys	304 SS 304L SS	0	A
		Nuts	304 SS	0	A
	Upper Support Plate Assembly	Bolts	316 SS	0	A
		Deep beam ribs	304 SS	1	A
		Deep beam stiffeners	304 SS	1	A

Table 7-2 (continued)
Categorization of Westinghouse Reactor Internals Components

Assembly	Subassembly	Component	Material	FMECA Group	Category
Upper Internals Assembly	Upper Support Plate Assembly	Flange	304 SS	0	A
		Inverted top hat (ITH) flange	304 SS	1	A
		Inverted top hat (ITH) upper support plate	304 SS	1	A
		Lock keys	316 SS	0	A
		Ribs	304 SS	0	A
		Upper support plate	304 SS	0	A
		Upper support ring or skirt	304 SS	2	B
Lower Internals Assembly	Baffle and Former Assembly	Baffle bolting lock bars	304 SS	1	A
		Baffle-edge bolts	316 SS 347 SS	3	C
		Baffle plates	304 SS	1	B
		Baffle-former bolts	316 SS 347 SS	2	C
		Barrel-former bolts	316 SS 347 SS	2	C
		Former plates	304 SS	1	B
	Bottom-Mounted Instrumentation (BMI) Column Assemblies	BMI column bodies	304 SS	1	B
		BMI column bolts	316 SS	1	A

Table 7-2 (continued)
Categorization of Westinghouse Reactor Internals Components

Assembly	Subassembly	Component	Material	FMECA Group	Category
Lower Internals Assembly	Bottom-Mounted Instrumentation (BMI) Column Assemblies	BMI column collars	304 SS	1	B
		BMI column cruciforms	CF8	1	B
		BMI column extension bars	304 SS	1	A
		BMI column extension tubes	304 SS	1	B
		BMI column lock caps	304L SS	0	A
		BMI column nuts	304 SS	1	A
	Core Barrel	Core barrel flange	304 SS	2	B
		Core barrel outlet nozzles	304 SS	2	B
		Lower core barrel	304 SS	3	C
		Upper core barrel	304 SS	3	C
	Diffuser Plate	Diffuser plate	304 SS	0	A
	Flux Thimbles (Tubes)	Flux thimble tube plugs	304 SS	1	B
		Flux thimbles (tubes)	316 SS	2	C
	Head Cooling Spray Nozzles	Head cooling spray nozzles	304 SS	0	A
	Irradiation Specimen Guides	Irradiation specimen guides	304 SS	1	A
		Irradiation specimen guide bolts	316 SS	1	A
		Irradiation specimen lock caps	304L SS	1	A

Table 7-2 (continued)
Categorization of Westinghouse Reactor Internals Components

Assembly	Subassembly	Component	Material	FMECA Group	Category
Lower Internals Assembly	Irradiation Specimen Guides	Specimen plugs	304 SS	1	A
	Lower Core Plate and Fuel Alignment Pins	Fuel alignment pins	316 SS	1	A
		LCP–fuel alignment pin bolts	316 SS	1	A
		LCP–fuel alignment pin lock caps	304L SS	1	A
		Lower core plate	304 SS	2	C
		XL lower core plate	304 SS	2	C
	Lower Support Column Assembly	Lower support column bodies	CF8	1	B
		Lower support column bodies	304 SS	1	B
		Lower support column bolts	304 SS	1	B
		Lower support column nuts	304 SS	0	A
		Lower support column sleeves	304 SS	0	A
	Lower Support Casting or Forging	Lower support casting	CF8	1	A
		Lower support forging	304 SS	0	A
	Neutron Panels/Thermal Shield	Neutron panel bolts	316 SS	1	A
		Neutron panel lock caps	304 SS	1	A
		Thermal shield bolts	316 SS	1	A
		Thermal shield dowels	316 SS	1	A

Table 7-2 (continued)
Categorization of Westinghouse Reactor Internals Components

Assembly	Subassembly	Component	Material	FMECA Group	Category
Lower Internals Assembly	Neutron Panels/Thermal Shield	Thermal shield flexures	304 SS	1	B
		Thermal shield or neutron panels	304 SS	1	A
	Radial Support Keys	Radial support key bolts	304 SS	1	A
		Radial support key lock keys	304 SS	0	A
		Radial support keys	304 SS	1	A
	Secondary Core Support (SCS) Assembly	SCS base plate	304 SS	1	A
		SCS bolts	316 SS	0	A
		SCS energy absorber	304 SS	0	A
		SCS guide post	304 SS	0	A
		SCS housing	304 SS	0	A
		SCS lock keys	304 SS	0	A
Interfacing Components	Interfacing Components	Clevis insert bolts	Alloy X-750	1	B
		Clevis insert lock keys	Alloy 600	0	A
		Clevis insert lock keys	316 SS	0	A
		Clevis inserts	Alloy 600	1	A
		Clevis inserts	304 SS	1	A

Table 7-2 (continued)
Categorization of Westinghouse Reactor Internals Components

Assembly	Subassembly	Component	Material	FMECA Group	Category
Interfacing Components	Interfacing Components	Clevis inserts	Stellite	1	A
		Head and vessel alignment pin bolts	316 SS	0	A
		Head and vessel alignment pin lock caps	304L SS	0	A
		Head and vessel alignment pins	304 SS	0	A
		Internals hold-down spring	304 SS	1	B
		Internals hold-down spring	403 SS	1	A
		Upper core plate alignment pins	304 SS	1	B

Table 7-3
Categorization of Combustion Engineering Reactor Internals Components

Assembly	Component	Material	FMECA Group	Category
Upper Internals Assembly	Upper guide structure support plate	304 SS	1	A
	Upper guide structure support flange–upper	304 SS	1	A
	Upper guide structure support flange–lower	304 SS	1	A
	Cylindrical skirt	304 SS	1	A
	Grid plate	304 SS	1	A
	Control rod shroud–grid ring	304 SS	1	A
	Control rod shroud–grid beams	304 SS	1	A
	Control rod shroud–cross braces	304 SS	1	A
	Control rod shroud–bolts	316 SS	0	A
	GSSS guide structure plate	304 SS	1	A
	GSSS support cylinder	304 SS	1	A
	GSSS studs	316 SS	0	A
	GSSS spherical washer sets	UNS S21800	0	A
	Flange blocks	304 SS	1	A
	Flange block bolts	410 SS	1	A
	Flange block shear pins	A286 SS	0	A
	RVLMS support structure tubes	304 SS	1	A

Table 7-3 (continued)
Categorization of Combustion Engineering Reactor Internals Components

Assembly	Component	Material	FMECA Group	Category
Upper Internals Assembly	Fuel alignment plate	304 SS	2	B
	Fuel bundle guide pins	316 SS	1	A
	Fuel bundle guide pin nuts	304 SS	1	A
	Hold-down ring	403 SS F6NM	1	A
	Belleville washer	Alloy 718	1	A
Lower Support Structure	Core support plate	304 SS 304L SS	2	C
	Core support plate bolts	316 SS	1	A
	Core support plate dowel pins	304 SS	1	A
	Anchor block bolts	316 SS	1	A
	Anchor block dowel pins	304 SS	1	A
	Fuel alignment pins	304 SS	1	A
	Fuel alignment pins	A286 SS	2	C
	Core support columns	304 SS	1	B
	Core support columns	CF8	1	B
	Core support beams	304 SS	1	A
	Core support deep beams	304 SS	2	C

Table 7-3 (continued)
Categorization of Combustion Engineering Reactor Internals Components

Assembly	Component	Material	FMECA Group	Category
Lower Support Structure	Core support column bolts	316 SS	1	B
	Bottom plate	304 SS	1	A
	ICI support columns	304 SS	1	A
Control Element Assembly (CEA)–Shroud Assemblies	CEA shrouds	304 SS	1	A
	CEA shrouds	CPF8 CF8	1	A
	CEA shrouds bases	304 SS	1	A
	CEA shrouds bases	CF8	1	A
	CEA shroud extension shaft guides	304 SS	1	A
	Modified CEA shroud extension shaft guides	CF8	1	A
	Instrument tubes	304 SS	1	B
	Internal/external spanner nuts	304 SS	1	A
	CEA shroud bolts	A286 SS	1	A
	CEA shroud tie rods	304 SS	1	A
	Snubber blocks	304 SS	1	A
	Snubber shims	XM-29	1	A
	Shim bolts	316 SS	0	A

Table 7-3 (continued)
Categorization of Combustion Engineering Reactor Internals Components

Assembly	Component	Material	FMECA Group	Category
Core Support Barrel Assembly	Upper cylinder	304 SS	2	B
	Lower cylinder	304 SS	3	C
	Upper core barrel flange	304 SS	2	B
	Lower core barrel flange	304 SS	2	B
	Core barrel snubber lugs	304 SS 321 SS 348 SS	1	A
	Core barrel snubber lug bolts	316 SS	0	A
	Core barrel snubber lug bolts	A286 SS	0	A
	Alignment keys	A286 SS	1	A
	Alignment keys	304 SS	1	A
	Alignment key dowel pins	304 SS	0	A
	Core barrel outlet nozzles	304 SS	1	A
	Thermal shield	304 SS	1	A
	Thermal shield positioning pins	UNS S21800	1	B
	Thermal shield support pins	304 SS	1	A
Core Shroud Assembly	Shroud plates	304 SS	2	C
	Former plates	304 SS	1	B

Table 7-3 (continued)
Categorization of Combustion Engineering Reactor Internals Components

Assembly	Component	Material	FMECA Group	Category
Core Shroud Assembly	Ribs	304 SS	1	B
	Rings	304 SS	1	B
	Core shroud bolts	316 SS	1	B
	Barrel-core shroud bolts	316 SS	1	B
	Core shroud tie rods	348 SS	1	B
	Core shroud tie rod nuts	316 SS	1	B
	Guide lugs	304 SS 348 SS	1	A
	Guide lug insert bolts	A286 SS	1	B
	Guide lug inserts	304 SS 321 SS 348 SS	1	A
In-Core Instrumentation (ICI)	ICI guide tubes	316 SS	1	A
	ICI nozzle support plate	304 SS	1	A
	ICI thimble support plate	304 SS	1	A
	ICI thimble tubes-upper	304 SS	1	A
	ICI thimble tubes-lower	Zircaloy-4	1	C

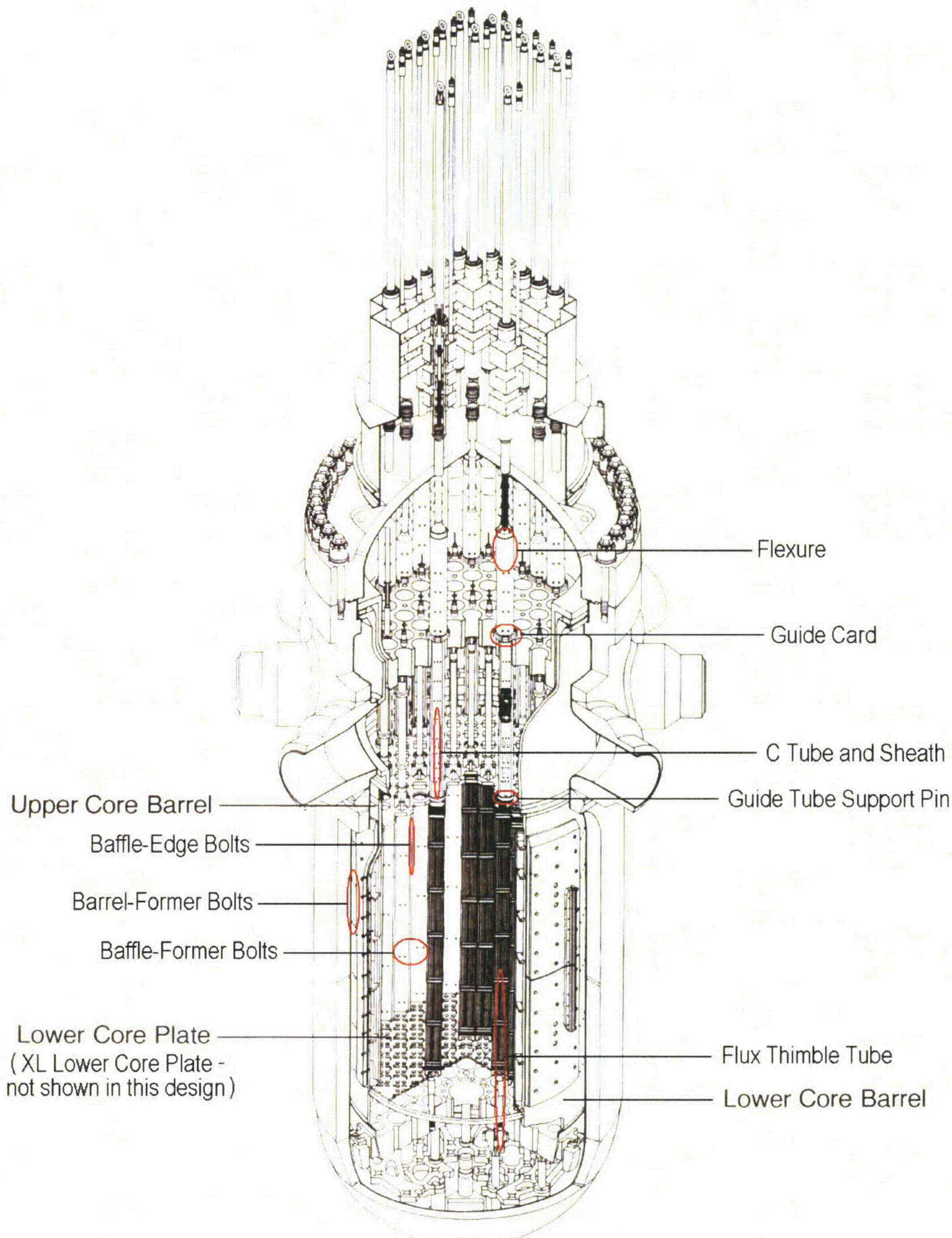


Figure 7-1
Location of Category C Components in Westinghouse Reactor Internals

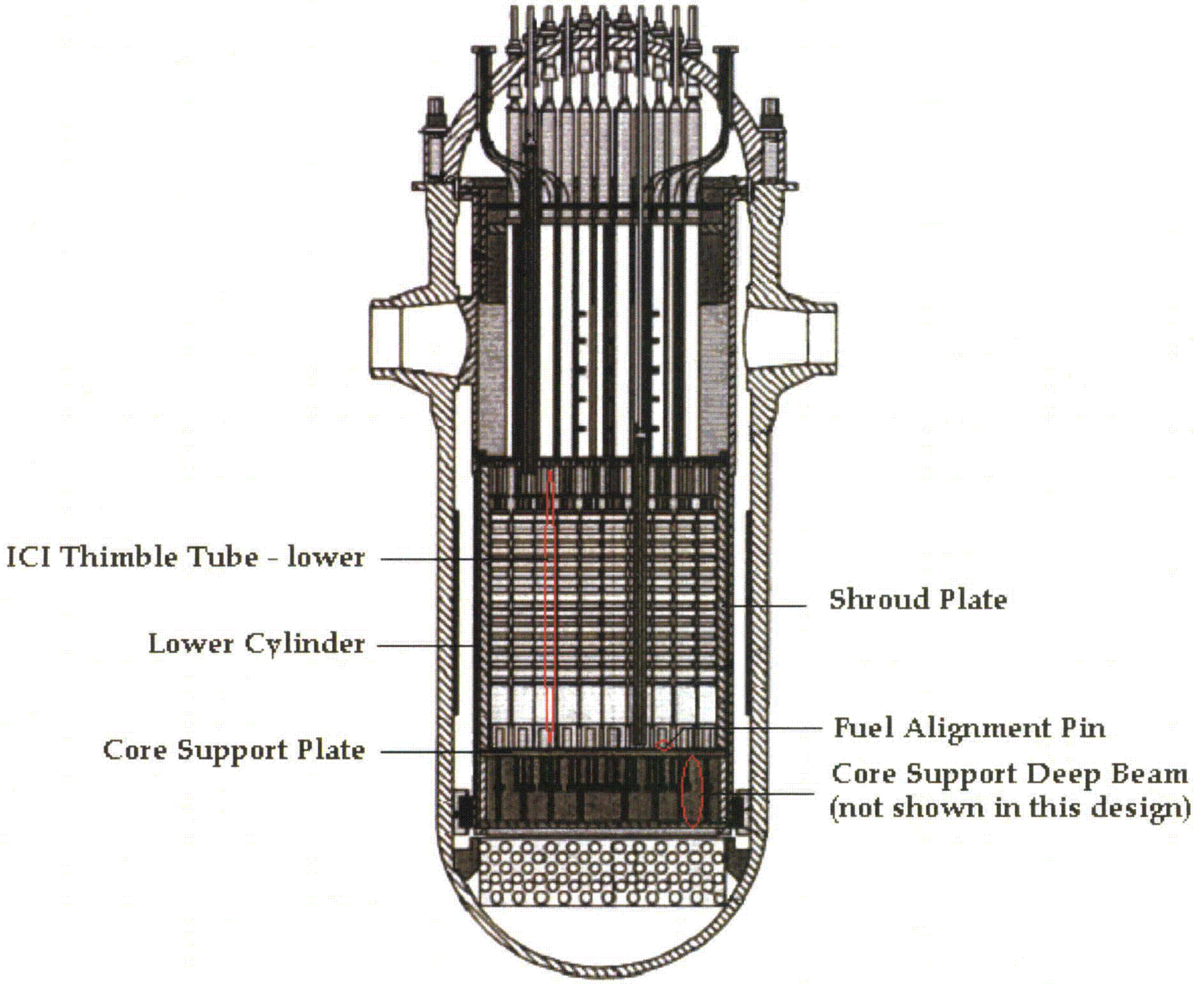


Figure 7-2
Location of Category C Components in CE Reactor Internals

8 CONCLUSIONS

This report provides the results of the reactor internals functionality analysis screening, categorization and ranking program for reactor internals components of Westinghouse and Combustion Engineering NSSS designs. This was performed in accordance with the framework and strategies for managing aging effects in PWR internals cited in MRP-134 [6]. The results of the categorization given in Section 7 are based on the culmination of several tasks as summarized below.

The process employed consisted of the following tasks:

- Data Compilation and Review
- Screening of Components
- Failure Modes, Effects and Criticality Analysis [FMECA]
- Categorization and Ranking

The process for each of these tasks is described in Section 2 with the results for each task in report Sections 4, 5, 6 and 7, respectively.

The Data Compilation and Review task assembled the information needed to conduct the screening of the reactor internals components. The specific input included identification of the age-related degradation mechanisms and compilation, characterization and grouping of the internals components for both Westinghouse and Combustion Engineering designs as described in Section 4. Those designs comprise 62 individual domestic plants. The screening of components with respect to the eight different age-related degradation mechanisms is described in Section 5. This process utilized the screening criteria from MRP-175 [4]. The screening process was used as an initial sorting method to place components into either Category A or Non-Category A. Screening results for the Westinghouse and Combustion Engineering NSSS designs are summarized in Tables 5-1 and 5-2.

Subsequent to the screening process, the Non-Category A components were evaluated in a Failure Modes, Effects and Criticality Analysis (FMECA). The results of this analysis are described in Section 6 and are summarized in Tables 6-5 and 6-6. Preliminary ratings for the significance of aging degradation compiled in the FMECA evaluation were used to determine those components that were likely candidates as Category C (Lead) items. Components that fell

into FMECA Groups 2 and 3 were then assessed for significance[1] and ranked, as detailed in Section 7, Table 7-1. Output from the screening process, the FMECA and the aging degradation significance evaluation were used to categorize all of the Westinghouse and Combustion Engineering NSSS designed components as shown in Tables 7-2 and 7-3.

Table 8-1 lists all of the Westinghouse and Combustion Engineering components identified as Category C. Figures 7-1 and 7-2 are provided to indicate the locations of each of the Category C components. Preliminary categories assigned to the components in this document may be subject to revision, pending the outcome of subsequent functionality assessments of this MRP program.

[1] Significance is identified as a "relative measure of aging degradation" in MRP-134 [6], with an equation to quantify its value for purposes of prioritizing the significance (ranking).

Table 8-1
Category C Components

Assembly	Components
Westinghouse-designed Internals	
Lower Internals	Baffle-former bolts
Lower Internals	Baffle-edge bolts
Lower Internals	Barrel-former bolts
Lower Internals	Lower core plate
Lower Internals	XL lower core plate
Upper Internals	Guide plates/cards
Upper Internals	Guide tube support pins (Alloy X-750)
Lower Internals	Lower core barrel
Lower Internals	Upper core barrel
Upper Internals	C tubes
Upper Internals	Sheaths
Lower Internals	Flux thimbles (tubes)
Upper Internals	Flexures
CE-designed Internals	
Core Shroud Assembly	Shroud plates
Lower Support Structure	Core support plate
Lower Support Structure	Core support deep beams
Lower Support Structure	Fuel alignment pins (A286SS)
Core Support Barrel Assembly	Lower cylinder
In-core Instrumentation	ICI thimble tubes

9 REFERENCES

1. *Pressurized Water Reactor Issue Management Tables, PWR-IMT Consequence of Failure (MRP-156)*, Palo Alto, CA, 2005, TP-1012110.
2. WCAP-14577, Revision 1A, License Renewal Evaluation: Aging Management for Reactor Internals, March 2001 (NRC approved, Non-Proprietary).
3. Generic Aging Management Review Report for the Reactor Internals, CE NPSD-1216, Rev. 0, March 2001.
4. *Materials Reliability Program: PWR Internals Material Aging Degradation Mechanism Screening and Threshold Values (MRP-175)*, EPRI, Palo Alto, CA: October 2005. 1012081.
5. WCAP-13627, Pilot Application of Risk-Based Inspection Methods to Westinghouse-Designed Reactor Internals, June 1993 (Non-Proprietary).
6. *Materials Reliability Program: Framework and Strategies for Managing Aging Effects in Reactor Internals (MRP-134)*, EPRI, Palo Alto, CA: June 2005. 1008203.
7. Screening Criteria for PWR Internals Components Ranking and Categorization, H. T. Tang, EPRI, Palo Alto, CA: November 1, 2005. MRP-2005-018.
8. RSICC Computer Code Collection CCC-650, "DOORS 3.1, One-, Two-, and Three-Dimensional Discrete Ordinates Neutron/Photon Transport Code System," Radiation Safety Information Computational Center, Oak Ridge National Laboratory (ORNL), August 1996.
9. RSICC Data Library Collection DLC-185, "BUGLE-96, Coupled 47 Neutron, 20 Gamma Ray Group Cross-Section Library Derived from ENDF/B-VI for LWR Shielding and Pressure Vessel Dosimetry Applications," Radiation Safety Information Computational Center, Oak Ridge National Laboratory (ORNL), March 1996.
10. "Safety Evaluation by the Office of Nuclear Reactor Regulation WCAP-14040, Revision 3, "Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and Reactor Coolant System Heatup and Cooldown Limit Curves" Westinghouse Owners Group Project No. 694, United States Nuclear Regulatory Commission, February 2004.
11. ANSI/IEEE, Standard 352-1987, "IEEE Guide for General Principles of Reliability Analysis of Nuclear Power Generating Station Safety Systems," IEEE, 1987.

A
SCREENING INPUT

This Appendix presents the results of the initial data collection process along with the output from interviews and discussions with internals designers and analysts.

The fluence regions identified in these tables indicate the maximum fluence experienced by a given component. Section 4 presents a description of the six fluence regions identified for the internals.

Table A-1
Results of Parameter Screening and Interviews with Analysts—Westinghouse Reactor Internals

Assembly/Component Name	Material Category	Material Type/Grade	Effective Stress ≥ Threshold	Structural Weld	Wear Potential	CUF > 0.1	Preload Req'd	Neutron Fluence Region
Upper Internals Assembly								
Control Rod Guide Tube Assemblies and Flow Downcomers								
Anti-rotation studs and nuts	Austenitic SS	304 SS	No	No	No	No	Yes	Region 1
Bolts	Austenitic SS	316 SS	No	No	No	No	Yes	Region 1
C tubes	Austenitic SS	304 SS	No	No	Yes	No	No	Region 3
Enclosure pins	Austenitic SS	304 SS	No	Yes	Yes	No	No	Region 1
Upper guide tube enclosures	Austenitic SS	304 SS	No	Yes	No	No	No	Region 1
Flanges-intermediate	Austenitic SS	304 SS	Yes	Yes	No	Yes	No	Region 1
Flanges-intermediate	Cast Austenitic SS	CF8	Yes	Yes	No	Yes	No	Region 1
Flanges-lower	Austenitic SS	304 SS	Yes	Yes	No	Yes	No	Region 3
Flanges-lower	Cast Austenitic SS	CF8	Yes	Yes	No	Yes	No	Region 3
Flexureless inserts	Austenitic SS	304 SS	No	No	No	No	No	Region 1
Flexures	PH Ni-base Alloy	Alloy X-750	Yes	No	No	No	Yes	Region 1
Guide plates/cards	Austenitic SS	304 SS	Yes	Yes	Yes	Yes	No	Region 1

Table A-1 (continued)
Results of Parameter Screening and Interviews with Analysts—Westinghouse Reactor Internals

Assembly/Component Name	Material Category	Material Type/Grade	Effective Stress ≥ Threshold	Structural Weld	Wear Potential	CUF > 0.1	Preload Req'd	Neutron Fluence Region
Control Rod Guide Tube Assemblies and Flow Downcomers								
Guide tube support pins	PH Ni-base Alloy	Alloy X-750	Yes	No	Yes	Yes	Yes	Region 3
Guide tube support pins	Austenitic SS	316 SS	Yes	No	Yes	Yes	Yes	Region 3
Housing plates	Austenitic SS	304 SS	No	No	No	No	No	Region 1
Inserts	Austenitic SS	304 SS	No	No	No	No	No	Region 1
Lock bars	Austenitic SS	304 SS	No	No	No	No	No	Region 1
Sheaths	Austenitic SS	304 SS	No	No	Yes	No	No	Region 3
Support pin cover plates	Austenitic SS	304 SS	No	No	No	No	No	Region 3
Support pin cover plate cap screws	Austenitic SS	316 SS	No	No	No	No	No	Region 3
Support pin cover plate locking caps and tie straps	Austenitic SS	304 SS	No	No	No	No	No	Region 3
Support pin nuts	PH Ni-base Alloy	Alloy X-750	No	No	No	No	No	Region 3
Support pin nuts	Austenitic SS	316 SS	No	No	No	No	No	Region 3
Water flow slot ligaments	Austenitic SS	304 SS	No	No	No	No	No	Region 3

Table A-1 (continued)
Results of Parameter Screening and Interviews with Analysts—Westinghouse Reactor Internals

Assembly/Component Name	Material Category	Material Type/Grade	Effective Stress ≥ Threshold	Structural Weld	Wear Potential	CUF > 0.1	Preload Req'd	Neutron Fluence Region
Mixing Devices								
Mixing devices	Cast Austenitic SS	CF8	No	Yes	No	No	No	Region 3
Upper Core Plate and Fuel Alignment Pins								
Fuel alignment pins	Austenitic SS	316 SS	Yes	No	Yes	No	No	Region 3
Upper core plate	Austenitic SS	304 SS	Yes	No	Yes	Yes	No	Region 3
Upper Instrumentation Conduit and Supports								
Bolting	Austenitic SS	316 SS	No	No	No	No	Yes	Region 1
Brackets, clamps, terminal blocks, and conduit straps	Austenitic SS	304 SS	No	No	No	No	No	Region 1
Conduit seal assembly–body, tubesheets	Austenitic SS	304 SS	No	No	No	No	No	Region 1
Conduit seal assembly–tubes	Austenitic SS	304 SS	Yes	No	No	No	No	Region 1
Conduits	Austenitic SS	304 SS	No	No	No	No	No	Region 1
Flange base	Austenitic SS	304 SS	No	No	No	No	No	Region 1
Locking caps	Austenitic SS	304 SS	No	No	No	No	No	Region 1
Support tubes	Austenitic SS	304 SS	No	No	No	No	No	Region 1

Table A-1 (continued)
Results of Parameter Screening and Interviews with Analysts—Westinghouse Reactor Internals

Assembly/Component Name	Material Category	Material Type/Grade	Effective Stress ≥ Threshold	Structural Weld	Wear Potential	CUF > 0.1	Preload Req'd	Neutron Fluence Region
Upper Plenum								
UHI flow column bases	Cast Austenitic SS	CF8	No	No	No	No	No	Region 3
UHI flow columns	Austenitic SS	304 SS	No	No	No	No	No	Region 3
Upper Support Column Assemblies								
Adapters	Austenitic SS	304 SS	No	No	No	No	No	Region 1
Bolts	Austenitic SS	316 SS	Yes	No	No	No	Yes	Region 3
Column bases	Cast Austenitic SS	CF8	Yes	No	No	No	No	Region 3
Column bodies	Austenitic SS	304 SS	Yes	No	No	No	No	Region 3
Extension tubes	Austenitic SS	304 SS	No	Yes	No	No	Yes	Region 1
Flanges	Austenitic SS	304 SS	No	No	No	No	No	Region 1
Lock keys	Austenitic SS	304 SS 304L SS	No	No	No	No	No	Region 1
Nuts	Austenitic SS	304 SS	No	No	No	No	No	Region 1
Upper Support Plate Assembly								
Bolts	Austenitic SS	316 SS	Yes	No	No	No	Yes	Region 3

Table A-1 (continued)
Results of Parameter Screening and Interviews with Analysts—Westinghouse Reactor Internals

Assembly/Component Name	Material Category	Material Type/Grade	Effective Stress ≥ Threshold	Structural Weld	Wear Potential	CUF > 0.1	Preload Req'd	Neutron Fluence Region
Upper Support Plate Assembly								
Deep beam ribs	Austenitic SS	304 SS	No	Yes	No	No	No	Region 1
Deep beam stiffeners	Austenitic SS	304 SS	No	Yes	No	No	No	Region 1
Flange	Austenitic SS	304 SS	No	No	No	No	No	Region 1
Inverted top hat (ITH) flange	Austenitic SS	304 SS	No	Yes	No	Yes	No	Region 1
Inverted top hat (ITH) upper support plate	Austenitic SS	304 SS	No	Yes	No	No	No	Region 1
Lock keys	Austenitic SS	316 SS	No	No	No	No	No	Region 1
Ribs	Austenitic SS	304 SS	No	No	No	No	No	Region 1
Upper support plate	Austenitic SS	304 SS	No	No	No	No	No	Region 1
Upper support ring or skirt	Austenitic SS	304 SS	Yes	Yes	No	Yes	No	Region 1
Lower Internals Assembly								
Baffle and Former Assembly								
Baffle bolting lock bars	Austenitic SS	304 SS	Yes	No	No	No	No	Region 6

Table A-1 (continued)
Results of Parameter Screening and Interviews with Analysts—Westinghouse Reactor Internals

Assembly/Component Name	Material Category	Material Type/Grade	Effective Stress ≥ Threshold	Structural Weld	Wear Potential	CUF > 0.1	Preload Req'd	Neutron Fluence Region
Baffle and Former Assembly								
Baffle-edge bolts	Austenitic SS	316 SS 347 SS	Yes	No	No	Yes	Yes	Region 6
Baffle plates	Austenitic SS	304 SS	No	No	No	No	No	Region 6
Baffle-former bolts	Austenitic SS	316 SS 347 SS	Yes	No	No	Yes	Yes	Region 6
Barrel-former bolts	Austenitic SS	316 SS 347 SS	Yes	No	No	Yes	Yes	Region 6
Former plates	Austenitic SS	304 SS	No	No	No	No	No	Region 6
Bottom Mounted Instrumentation (BMI) Column Assemblies								
BMI column bodies	Austenitic SS	304 SS	Yes	Yes	No	Yes	No	Region 5
BMI column bolts	Austenitic SS	316 SS	Yes	No	No	Yes	Yes	Region 1
BMI column collars	Austenitic SS	304 SS	No	No	No	No	No	Region 5
BMI column cruciforms	Cast Austenitic SS	CF8	No	No	No	No	No	Region 5
BMI column extension bars	Austenitic SS	304 SS	No	No	No	No	No	Region 5
BMI column extension tubes	Austenitic SS	304 SS	Yes	Yes	No	Yes	No	Region 5
BMI column lock caps	Austenitic SS	304L SS	No	No	No	No	No	Region 1

Table A-1 (continued)
Results of Parameter Screening and Interviews with Analysts—Westinghouse Reactor Internals

Assembly/Component Name	Material Category	Material Type/Grade	Effective Stress ≥ Threshold	Structural Weld	Wear Potential	CUF > 0.1	Preload Req'd	Neutron Fluence Region
Bottom Mounted Instrumentation (BMI) Column Assemblies								
BMI column nuts	Austenitic SS	304 SS	No	No	No	No	Yes	Region 5
Core Barrel								
Core barrel flange	Austenitic SS	304 SS	No	Yes	Yes	No	No	Region 1
Core barrel outlet nozzles	Austenitic SS	304 SS	Yes	Yes	No	Yes	No	Region 1
Lower core barrel	Austenitic SS	304 SS	Yes	Yes	No	No	No	Region 4
Upper core barrel	Austenitic SS	304 SS	Yes	Yes	No	No	No	Region 4
Diffuser Plate								
Diffuser plate	Austenitic SS	304 SS	No	No	No	No	No	Region 1
Flux Thimbles (Tubes)								
Flux thimble tube plugs	Austenitic SS	304 SS	No	Yes	No	No	No	Region 5
Flux thimbles (tubes)	Austenitic SS	316 SS	No	Yes	Yes	No	No	Region 5
Head Cooling Spray Nozzles								
Head cooling spray nozzles	Austenitic SS	304 SS	No	No	No	No	No	Region 1

Table A-1 (continued)
Results of Parameter Screening and Interviews with Analysts—Westinghouse Reactor Internals

Assembly/Component Name	Material Category	Material Type/Grade	Effective Stress ≥ Threshold	Structural Weld	Wear Potential	CUF > 0.1	Preload Req'd	Neutron Fluence Region
Irradiation Specimen Guides								
Irradiation specimen guides	Austenitic SS	304 SS	No	No	Yes	No	No	Region 4
Irradiation specimen guide bolts	Austenitic SS	316 SS	Yes	No	No	Yes	Yes	Region 4
Irradiation specimen guide lock caps	Austenitic SS	304L SS	No	No	No	No	No	Region 4
Specimen plugs	Austenitic SS	304 SS	No	No	No	No	No	Region 4
Lower Core Plate and Fuel Alignment Pins								
Fuel alignment pins	Austenitic SS	316 SS	Yes	No	Yes	No	No	Region 5
LCP-fuel alignment pin bolts	Austenitic SS	316 SS	Yes	No	Yes	No	Yes	Region 5
LCP-fuel alignment pin lock caps	Austenitic SS	304L SS	No	No	No	No	No	Region 5
Lower core plate	Austenitic SS	304 SS	Yes	Yes	Yes	Yes	No	Region 5
XL lower core plate	Austenitic SS	304 SS	Yes	Yes	Yes	Yes	No	Region 4
Lower Support Column Assemblies								
Lower support column bodies	Cast Austenitic SS	CF8	No	No	No	No	No	Region 5

Table A-1 (continued)
Results of Parameter Screening and Interviews with Analysts—Westinghouse Reactor Internals

Assembly/Component Name	Material Category	Material Type/Grade	Effective Stress ≥ Threshold	Structural Weld	Wear Potential	CUF > 0.1	Preload Req'd	Neutron Fluence Region
Lower Support Column Assemblies								
Lower support column bodies	Austenitic SS	304 SS	No	No	No	No	No	Region 5
Lower support column bolts	Austenitic SS	304 SS	Yes	No	No	Yes	Yes	Region 5
Lower support column nuts	Austenitic SS	304 SS	Yes	No	No	No	Yes	Region 1
Lower support column sleeves	Austenitic SS	304 SS	No	No	No	No	No	Region 1
Lower Support Casting or Forging								
Lower support casting	Cast Austenitic SS	CF8	No	No	No	No	No	Region 1
Lower support forging	Austenitic SS	304 SS	No	No	No	No	No	Region 1
Neutron Panels/Thermal Shield								
Neutron panel bolts	Austenitic SS	316 SS	Yes	No	No	No	Yes	Region 4
Neutron panel lock caps	Austenitic SS	304 SS	No	No	No	No	No	Region 4
Thermal shield bolts	Austenitic SS	316 SS	Yes	No	No	Yes	Yes	Region 4
Thermal shield dowels	Austenitic SS	316 SS	No	No	No	No	No	Region 4

Table A-1 (continued)
Results of Parameter Screening and Interviews with Analysts—Westinghouse Reactor Internals

Assembly/Component Name	Material Category	Material Type/Grade	Effective Stress ≥ Threshold	Structural Weld	Wear Potential	CUF > 0.1	Preload Req'd	Neutron Fluence Region
Neutron Panels/Thermal Shield								
Thermal shield flexures	Austenitic SS	304 SS	Yes	No	No	Yes	Yes	Region 4
Thermal shield or neutron panels	Austenitic SS	304 SS	No	No	No	No	No	Region 4
Radial Support Keys								
Radial support key bolts	Austenitic SS	304 SS	Yes	No	Yes	No	Yes	Region 1
Radial support key lock keys	Austenitic SS	304 SS	No	No	No	No	No	Region 1
Radial support keys	Austenitic SS	304 SS	No	Yes	Yes	No	No	Region 1
Secondary Core Support (SCS) Assembly								
SCS base plate	Austenitic SS	304 SS	No	Yes	No	No	No	Region 1
SCS bolts	Austenitic SS	316 SS	No	No	No	No	Yes	Region 1
SCS energy absorber	Austenitic SS	304 SS	No	No	No	No	No	Region 1
SCS guide post	Austenitic SS	304 SS	No	No	No	No	No	Region 1
SCS housing	Austenitic SS	304 SS	No	No	No	No	No	Region 1
SCS lock keys	Austenitic SS	304 SS	No	No	No	No	No	Region 1

Table A-1 (continued)
Results of Parameter Screening and Interviews with Analysts—Westinghouse Reactor Internals

Assembly/Component Name	Material Category	Material Type/Grade	Effective Stress ≥ Threshold	Structural Weld	Wear Potential	CUF > 0.1	Preload Req'd	Neutron Fluence Region
Interfacing Components								
Clevis insert bolts	PH Ni-base Alloy	Alloy X-750	Yes	No	Yes	No	Yes	Region 1
Clevis insert lock keys	Austenitic Ni-base Alloy	Alloy 600	No	No	No	No	No	Region 1
Clevis insert lock keys	Austenitic SS	316 SS	No	No	No	No	No	Region 1
Clevis inserts	Austenitic Ni-base Alloy	Alloy 600	No	No	Yes	No	No	Region 1
Clevis inserts	Austenitic SS	304 SS	No	No	Yes	No	No	Region 1
Clevis inserts	Co Hard Facing Alloy	Stellite	No	No	Yes	No	No	Region 1
Head and vessel alignment pin bolts	Austenitic SS	316 SS	No	No	No	No	Yes	Region 1
Head and vessel alignment pin lock caps	Austenitic SS	304L SS	No	No	No	No	No	Region 1
Head and vessel alignment pins	Austenitic SS	304 SS	No	No	No	No	No	Region 1
Internals hold-down spring	Austenitic SS	304 SS	Yes	No	Yes	No	Yes	Region 1
Internals hold-down spring	Martensitic SS	403 SS	No	No	Yes	No	Yes	Region 1
Upper core plate alignment pins	Austenitic SS	304 SS	No	Yes	Yes	No	No	Region 3

Table A-2
Results of Parameter Screening and Interviews with Analysts—Combustion Engineering Reactor Internals

Assembly/Component Name	Material Category	Material Type/Grade	Effective Stress ≥ Threshold	Structural Weld	Wear Potential	CUF > 0.1	Preload Req'd	Neutron Fluence Region
Upper Internals Assembly								
Upper guide structure support plate	Austenitic SS	304 SS	Yes	Yes	No	No	No	Region 1
Upper guide structure support flange-upper	Austenitic SS	304 SS	Yes	Yes	Yes	No	No	Region 1
Upper guide structure support flange-lower	Austenitic SS	304 SS	Yes	Yes	No	No	No	Region 1
Cylindrical skirt	Austenitic SS	304 SS	No	Yes	No	No	No	Region 1
Grid plate	Austenitic SS	304 SS	No	Yes	No	No	No	Region 1
Control rod shroud-grid ring	Austenitic SS	304 SS	No	Yes	No	No	No	Region 1
Control rod shroud-grid beams	Austenitic SS	304 SS	No	Yes	No	No	No	Region 1
Control rod shroud-cross braces	Austenitic SS	304 SS	No	Yes	No	No	No	Region 1
Control rod shroud-bolts	Austenitic SS	316 SS	No	No	No	No	Yes	Region 1
GSSS guide structure plate	Austenitic SS	304 SS	No	Yes	No	No	No	Region 1
GSSS support cylinder	Austenitic SS	304 SS	No	Yes	No	No	No	Region 1
GSSS studs	Austenitic SS	316 SS	No	No	No	No	Yes	Region 1

Table A-2 (continued)
Results of Parameter Screening and Interviews with Analysts—Combustion Engineering Reactor Internals

Assembly/Component Name	Material Category	Material Type/Grade	Effective Stress ≥ Threshold	Structural Weld	Wear Potential	CUF > 0.1	Preload Req'd	Neutron Fluence Region
Upper Internals Assembly								
GSSS spherical washer sets	Austenitic SS	UNS S21800	No	No	No	No	No	Region 1
Flange blocks	Austenitic SS	304 SS	No	No	Yes	No	No	Region 1
Flange block bolts	Martensitic SS	410 SS	No	No	No	No	Yes	Region 1
Flange block shear pins	PH Austenitic SS	A286 SS	No	No	No	No	No	Region 1
RVLMS support structure tubes	Austenitic SS	304 SS	No	Yes	Yes	Yes	Yes	Region 2
Fuel alignment plate	Austenitic SS	304 SS	Yes	Yes	Yes	Yes	No	Region 2
Fuel bundle guide pins	Austenitic SS	316 SS	No	No	Yes	No	Yes	Region 2
Fuel bundle guide pin nuts	Austenitic SS	304 SS	No	No	No	No	Yes	Region 2
Hold down ring	Martensitic SS	403 SS F6NM	No	No	Yes	No	Yes	Region 1
Belleville washer	PH Ni-base Alloy	Alloy 718	No	No	Yes	No	Yes	Region 1
Lower Support Structure								
Core support plate	Austenitic SS	304 SS 304L SS	Yes	Yes	Yes	Yes	No	Region 4
Core support plate bolts	Austenitic SS	316 SS	Yes	No	No	Yes	Yes	Region 4

Table A-2 (continued)
Results of Parameter Screening and Interviews with Analysts—Combustion Engineering Reactor Internals

Assembly/Component Name	Material Category	Material Type/Grade	Effective Stress ≥ Threshold	Structural Weld	Wear Potential	CUF > 0.1	Preload Req'd	Neutron Fluence Region
Lower Support Structure								
Core support plate dowel pins	Austenitic SS	304 SS	No	No	No	No	No	Region 4
Anchor block bolts	Austenitic SS	316 SS	No	No	No	No	Yes	Region 4
Anchor block dowel pins	Austenitic SS	304 SS	No	No	No	No	No	Region 4
Fuel alignment pins	Austenitic SS	304 SS	No	No	No	No	No	Region 4
Fuel alignment pins	PH Austenitic SS	A286 SS	No	No	No	No	Yes	Region 4
Core support columns	Austenitic SS	304 SS	Yes	Yes	No	Yes	No	Region 4
Core support columns	Cast Austenitic SS	CF8	Yes	Yes	No	Yes	No	Region 4
Core support beams	Austenitic SS	304 SS	Yes	Yes	No	Yes	No	Region 1
Core support deep beams	Austenitic SS	304 SS	Yes	Yes	No	Yes	No	Region 4
Core support column bolts	Austenitic SS	316 SS	Yes	No	No	No	Yes	Region 4
Bottom plate	Austenitic SS	304 SS	No	Yes	No	No	No	Region 1
ICI support columns	Austenitic SS	304 SS	No	Yes	No	No	No	Region 1

Table A-2 (continued)
Results of Parameter Screening and Interviews with Analysts—Combustion Engineering Reactor Internals

Assembly/Component Name	Material Category	Material Type/Grade	Effective Stress ≥ Threshold	Structural Weld	Wear Potential	CUF > 0.1	Preload Req'd	Neutron Fluence Region
Control Element Assembly (CEA) – Shroud Assemblies								
CEA shrouds	Austenitic SS	304 SS	No	Yes	No	No	No	Region 1
CEA shrouds	Cast Austenitic SS	CPF8 CF8	No	Yes	No	No	No	Region 1
CEA shroud bases	Austenitic SS	304 SS	No	Yes	No	No	No	Region 2
CEA shroud bases	Cast Austenitic SS	CF8	No	Yes	No	No	No	Region 2
CEA shroud extension shaft guides	Austenitic SS	304 SS	No	Yes	No	No	No	Region 1
Modified CEA shroud extension shaft guides	Cast Austenitic SS	CF8	No	Yes	No	No	No	Region 1
Instrument tubes	Austenitic SS	304 SS	Yes	Yes	No	Yes	No	Region 2
Internal/external spanner nuts	Austenitic SS	304 SS	No	Yes	No	No	Yes	Region 1
CEA shroud bolts	PH Austenitic SS	A286 SS	No	No	No	No	Yes	Region 2
CEA shroud tie rods	Austenitic SS	304 SS	No	Yes	No	No	Yes	Region 1
Snubber blocks	Austenitic SS	304 SS	No	Yes	No	No	No	Region 1

Table A-2 (continued)
Results of Parameter Screening and Interviews with Analysts—Combustion Engineering Reactor Internals

Assembly/Component Name	Material Category	Material Type/Grade	Effective Stress ≥ Threshold	Structural Weld	Wear Potential	CUF > 0.1	Preload Req'd	Neutron Fluence Region
Control Element Assembly (CEA) – Shroud Assemblies								
Snubber shims	Austenitic SS	XM-29	No	No	Yes	No	No	Region 1
Shim bolts	Austenitic SS	316 SS	No	No	No	No	Yes	Region 1
Core Support Barrel Assembly								
Upper cylinder	Austenitic SS	304 SS	No	Yes	No	No	No	Region 2
Lower cylinder	Austenitic SS	304 SS	Yes	Yes	No	No	No	Region 4
Upper core barrel flange	Austenitic SS	304 SS	Yes	Yes	Yes	No	No	Region 1
Lower core barrel flange	Austenitic SS	304 SS	No	Yes	No	Yes	No	Region 1
Core barrel snubber lugs	Austenitic SS	304 SS 321 SS 348 SS	No	Yes	Yes	No	No	Region 3
Core barrel snubber lug bolts	Austenitic SS	316 SS	No	No	No	No	Yes	Region 1
Core barrel snubber lug bolts	PH Austenitic SS	A286 SS	No	No	No	No	Yes	Region 1
Alignment keys	PH Austenitic SS	A286 SS	No	No	Yes	No	No	Region 1
Alignment keys	Austenitic SS	304 SS	No	No	Yes	No	No	Region 1

Table A-2 (continued)
Results of Parameter Screening and Interviews with Analysts—Combustion Engineering Reactor Internals

Assembly/Component Name	Material Category	Material Type/Grade	Effective Stress ≥ Threshold	Structural Weld	Wear Potential	CUF > 0.1	Preload Req'd	Neutron Fluence Region
Core Support Barrel Assembly								
Alignment key dowel pins	Austenitic SS	304 SS	No	No	No	No	No	Region 1
Core barrel outlet nozzles	Austenitic SS	304 SS	No	Yes	Yes	No	No	Region 2
Thermal shield	Austenitic SS	304 SS	No	Yes	No	No	No	Region 2
Thermal shield positioning pins	Austenitic SS	UNS S21800	No	No	Yes	No	Yes	Region 2
Thermal shield support pins	Austenitic SS	304 SS	No	No	Yes	No	No	Region 2
Core Shroud Assembly								
Shroud plates	Austenitic SS	304 SS	Yes	Yes	No	No	No	Region 5
Former plates	Austenitic SS	304 SS	Yes	Yes	No	No	No	Region 5
Ribs	Austenitic SS	304 SS	Yes	Yes	No	No	No	Region 5
Rings	Austenitic SS	304 SS	No	Yes	No	No	No	Region 5
Core shroud bolts	Austenitic SS	316 SS	Yes	No	No	Yes	Yes	Region 5
Barrel-core shroud bolts	Austenitic SS	316 SS	Yes	No	No	No	Yes	Region 4
Core shroud tie rods	Austenitic SS	348 SS	No	No	Yes	No	Yes	Region 4

Table A-2 (continued)
Results of Parameter Screening and Interviews with Analysts—Combustion Engineering Reactor Internals

Assembly/Component Name	Material Category	Material Type/Grade	Effective Stress ≥ Threshold	Structural Weld	Wear Potential	CUF > 0.1	Preload Req'd	Neutron Fluence Region
Core Shroud Assembly								
Core shroud tie rod nuts	Austenitic SS	316 SS	No	No	No	No	Yes	Region 4
Guide lugs	Austenitic SS	304 SS 348 SS	No	Yes	No	No	No	Region 2
Guide lug insert bolts	PH Austenitic SS	A286 SS	No	No	No	No	Yes	Region 2
Guide lug inserts	Austenitic SS	304 SS 321 SS 348 SS	No	No	Yes	No	No	Region 2
In-Core Instrumentation (ICI)								
ICI guide tubes	Austenitic SS	316 SS	No	Yes	No	No	No	Region 4
ICI nozzle support plate	Austenitic SS	304 SS	No	Yes	No	No	No	Region 1
ICI thimble support plate	Austenitic SS	304 SS	No	Yes	Yes	No	No	Region 1
ICI thimble tubes-upper	Austenitic SS	304 SS	No	Yes	Yes	No	No	Region 2
ICI thimble tubes-lower	Zirconium-base Alloy	Zircaloy-4	No	Yes	Yes	No	No	Region 6

B THERMAL STRESS RELAXATION IN AUSTENITIC STAINLESS STEELS

This Appendix presents a position agreed to by the Westinghouse expert panel involved with the review of screening results as part of the FMECA evaluation. It provides the technical basis for Westinghouse's position to eliminate thermal stress relaxation as a degradation mechanism to be evaluated for internals components at normal PWR operating temperatures. Wear and fatigue, both degradation mechanisms that may be linked to thermal stress relaxation according to the screening criteria, were likewise removed from consideration if thermal stress relaxation was eliminated.

B.1 The Issue

As part of the Materials Reliability Program PWR materials aging program evaluations, consideration is given to the potential consequences of a large number of degradation mechanisms that might lead to failure, or might otherwise compromise the reliability of PWR internals components during extended (to 60 years) service. These mechanisms include SCC, IASCC, wear, fatigue, and various other mechanisms that might lead to a "reduction in toughness" of the materials of construction. The latter rather broad grouping includes such effects as thermal embrittlement, radiation embrittlement, void swelling, irradiation-induced stress relaxation/irradiation creep and thermal stress relaxation.

In order to make the overall assessment tractable, a process has been established whereby: a) the specific materials for each component are identified, b) the stress, thermal, and radiation environments affecting each component are determined, and c) the latter are weighed against a set of "Screening Criteria" established by an expert solicitation process [B1, B2]. If the conditions for a given component or subassembly exceed the Screening Criteria established for one or more of the degradation mechanisms, that component or subassembly is then "screened in" and evaluated further.

Any rational process in which subjective judgment is required is not without flaws. For example, for a small number of cases, lacking rigorously applicable experimental data, the decision has been taken to "assume" a failure mechanism is plausible with the expectation that more detailed assessments may lead to its later elimination. For the current stage of this evaluation, it has been assumed that any component such as a bolt or spring in which the intended function requires application of a preload, will be subject to ***thermal stress relaxation*** resulting in loss of preload.

This decision then leads to the further argument that such components are now "loose" and, hence, subject also to ***wear*** and ***fatigue*** as consequential degradation mechanisms.

The purpose of this Appendix is to examine the validity of this position.

B.2 The Materials and Applications

There are several components in the Westinghouse and CE internals FMECA risk-based tables for which thermal stress relaxation (and wear, fatigue) has been screened in as a potential degradation mechanism. The materials in question are invariably one or more of the grades of austenitic stainless steel, typically Type 304 or Type 316. The service conditions include operation in the temperature range of approximately 290-320°C (554-608°F), very low or no neutron irradiation (60-year damage levels less than 0.2 dpa), and (elastic) mechanical preloads estimated as up to several hundred pounds.

The potential concern arises by consideration of what would happen if the preload was sufficiently relaxed to permit a bolt to become "loose" or a spring to become less resilient, thereby compromising its essential function. It is acknowledged that such stress relaxation can occur in highly irradiated components such as baffle former bolts; the related mechanism of irradiation creep is also known to occur in structural components subjected to high levels of irradiation damage. These latter components are exempted from this discussion.

B.3 The Mechanism of Stress Relaxation

Thermal stress relaxation and thermal creep are manifestations of the same phenomenological process. Whereas the former results in the decrease in stress under constant deformation, the latter occurs by combined elastic and plastic deformation under conditions of constant load. Unlike the result of constant strain-rate deformation–as, e.g., in a tensile test–the amount and rate of straining during creep are established by the material itself under the imposed stress and temperature conditions.

In a simple form, the total strain of such a stressed component can be expressed as [B3],

$$\varepsilon = \varepsilon_e + \varepsilon_p = \frac{\sigma}{E} + \varepsilon_p$$ **Equation B-1**

where,

ε = total strain

ε_e = elastic strain

ε_p = plastic strain

For the process of stress relaxation, for the total strain to remain constant as the material "creeps," the elastic strain must decrease. A particularly common example of this process is the reduction in residual stress that occurs by an action such as post-weld heat treatment, whereby the strain arising from the weld pool solidification process can be mitigated. Depending on temperature, metallurgical changes may also occur. Without the constraint of fixed deformation, the material may experience a large plastic (creep) strain which can dominate the strain accumulation process.

The relationship expressed by Equation B-1 is valid at any temperature. However, the extent to which plastic deformation can occur, and contribute to the elastic unloading process, is very strongly temperature-dependent [B4,B5].

The two significant stress-aided and thermally activated recovery processes in crystals are cross-slip and dislocation climb. At "lower temperatures" the cross-slip of screw dislocations is the only process by which obstacles in the slip plane can be bypassed or dislocations annihilated; except for isolated jogs, the applied forces necessary to make a dislocation "climb" rapidly approach the theoretical elastic limit [B5]. At elevated temperatures, where vacancy concentrations are much higher, creep can occur by diffusion-assisted dislocation climb. Most theories of high temperature creep are based on climb models which involve classical rate theory to recognize the role of thermal excitation. The temperature at which dislocation climb is normally expected to dominate the creep process is ~ 0.5 T_m (T_m is the absolute melting temperature). For austenitic stainless steels 0.5 T_m is approximately 600°C (~1100°F).

Austenitic stainless steels are characterized by relatively low stacking fault energies (SFEs). The consequence of this is an abundance of extended or partial dislocations and annealing twins in the microstructure. At low temperatures–e.g. below about 0.35 T_m–these features seriously impede dislocation glide on the normal slip planes, necessitating the cross-slip mentioned above to effect the recovery process.

Figure B-1 presents a schematic representation of creep under constant load [B6]. In the more specific case of stress relaxation, where the deformation is fixed and the stress is allowed to decay, the resulting decrease in stress is due solely to the transient or primary portion of the creep curve. At low temperatures, such as we are concerned with here, a plot of the stress relaxation as a function of time may be viewed as more or less an inverse form of the low temperature primary creep strain curve in the lower portion of Figure B-1. This is precisely why stress relaxation-time curves exhibit an asymptotic or saturation appearance.

B.4 Discussion and Conclusions

The internals components of interest operate in the temperature range corresponding to 0.3-0.33T_m. In this range, some relaxation of the short range elastic stresses will occur. Data presented in Ref. B1, however, show that the relaxation may be on the order of 10-20% in a few thousand hours, with further exposure time essentially showing saturation, with little subsequent additional decrease.

There is no physical basis to argue that significant further stress relaxation will occur at longer times. Once the initial "transient" period has been completed, the stress that is being "relaxed" decreases as the relaxation process proceeds. Thus, at any incremental time into the relaxation process the stress available is decreasing, leading to the observed saturation.

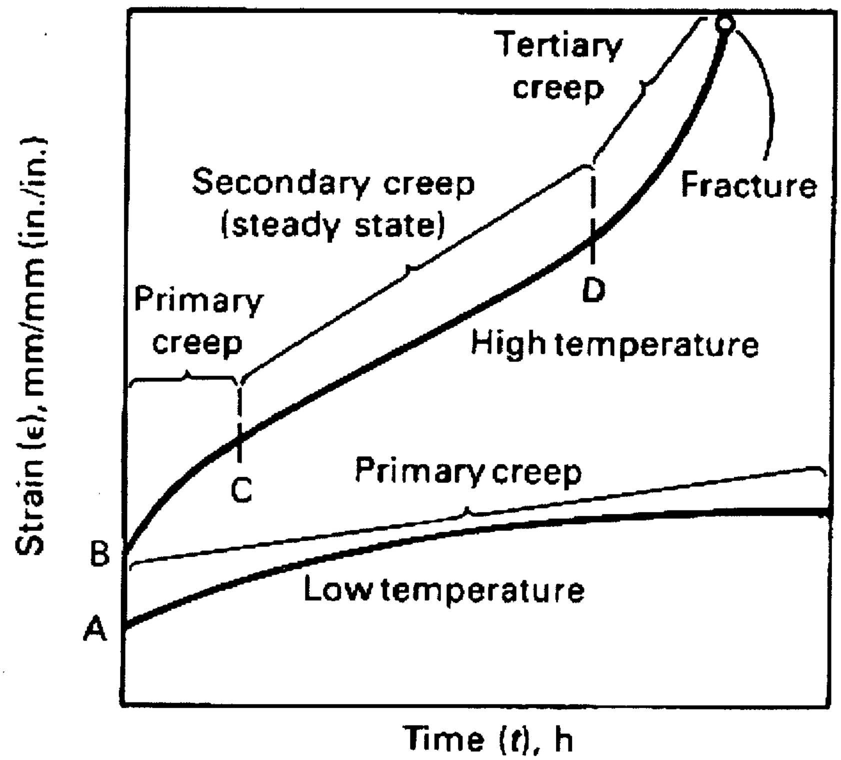


Figure B-1
Schematic Illustration of Thermal Creep as a Function of Temperature

From an examination of the physical processes involved, there is no apparent defendable evidence for assigning thermal stress relaxation as a relevant degradation mechanism for austenitic stainless steels operating in non-radiation environments at PWR temperatures. It is noteworthy to point out such an argument is also applicable to Alloy X-750 or Alloy 718 preloaded bolts or ferritic-martensitic stainless steel bolts or springs. Designations in the FMECA tables for both Westinghouse and CE internals were removed; where "wear" and "fatigue" were screened in as secondary mechanisms, these were also removed.

It is worth observing that the logic represented in this Appendix is consistent with the discussion of creep and stress relaxation, in the absence of a significant radiation environment, found in NUREG/CR-6048 [B7].

B.5 Appendix B References

B1. *Materials Reliability Program: PWR Internals Material Aging Degradation Mechanism Screening and Threshold Values (MRP-175)*, EPRI, Palo Alto, CA: October 2005. 1012081.

B2. *Screening Criteria for PWR Internals Components Ranking and Categorization*, H. T. Tang, EPRI, Palo Alto, CA: November 1, 2005. MRP-2005-018.

B3. G. E. Dieter, Mechanical Metallurgy, McGraw-Hill (1961).

B4. The Structure and Properties of Materials, Vol. III, Mechanical Behavior, H. W. Hayden et al., John Wiley & Sons (1965).

B5. J. Friedel, Dislocations, Addison-Wesley Publishing (1964).

B6. *Materials Reliability Program, Technical Basis Document Concerning Irradiation Induced Stress Relaxation and Void Swelling in PWR RV Internals Components (MRP-50)*, EPRI, Palo Alto, CA: 2001. 1000970.

B7. *Pressurized Water Reactor Internals Aging Degradation Study, Phase I*, NUREG/CR-6048, Sept. 1993.

C
COMPONENT FUNCTIONAL DESCRIPTIONS FOR WESTINGHOUSE REACTOR INTERNALS

The Westinghouse reactor internals assembly is part of the reactor coolant system and is located inside the reactor pressure vessel. The reactor internals are long-lived passive structural components. The intended functions are to support core cooling, enable control rod insertion, and maintain the integrity of the fuel. Internals components are classified as either core support structures or internals structures. The components are grouped under the upper internals assembly and the lower internals assembly.

Components and Functions Overview of the Westinghouse Pressurized Water Reactor Internals

The reactor internals perform the following intended functions:

- Provide the capability to shut down the reactor and maintain it in a safe shutdown condition
- Prevent failure of all non-safety-related systems, structures, and components whose failure could prevent any of these functions

In addition, since the bottom-mounted flux thimbles are included, there is an additional intended function, to ensure that the integrity of the reactor coolant pressure boundary is maintained. (The flux thimbles are the only pressure boundary component included here).

The individual subcomponents comprising the reactor vessel internals provide the following services:

- Support and orientation of the reactor core (that is, the fuel assemblies)
- Support, orientation, guidance, and protection of the control rod assemblies
- A passageway for the distribution of the reactor coolant flow to the reactor core
- A passageway for support, guidance, and protection for in-core instrumentation
- A secondary core support for limiting the core support structure downward displacement
- Gamma and neutron shielding for the reactor pressure vessel

All Westinghouse reactor internals consist of two basic assemblies: an upper internals assembly that is removed during each refueling operation to obtain access to the reactor core and a lower internals assembly that can be removed following a complete core unload. The fact that all of the

internals can be removed from the reactor vessel ensures the capability to perform periodic inspections to determine the condition of the internals or to effect repairs, if needed.

The lower internals assembly is supported in the vessel by clamping to a ledge below the vessel-head mating surface and is closely guided at the bottom by radial support/clevis assemblies. The upper internals assembly is clamped at this same ledge by the reactor vessel head. The bottom of the upper internals assembly is closely guided by the core barrel alignment pins of the lower internals assembly.

There are three basic models of reactor internals in domestic Westinghouse Owners Group plants: two-loop, three-loop, and four-loop. There are variations within these categories, such as, core lengths, ratings, support geometry, and material product forms. A designation can be made relative to the design of the upper support plate assemblies, that is, 1) a deep beam model, 2) a top hat model, or 3) an inverted top hat model.

During reactor operation, the upper internals assembly is preloaded against the fuel assembly springs and the internals hold-down spring by the reactor vessel head pressing down on the outside edge of the upper support plate. The upper support plate acts as the divider between the upper plenum and the upper head and as a rigid base for the rest of the upper internals components. From the upper support plate, the upper support columns and the guide tubes are attached. The upper core plate (UCP), in turn, is attached to the upper support columns.

The UCP is perforated to permit coolant to pass from the core into the upper plenum defined by the upper support plate and the UCP. The coolant then exits through the outlet nozzles in the core barrel. The UCP positions and laterally supports the core by fuel pins extending below the plate. The UCP contacts and preloads the fuel assembly springs and, therefore, maintains contact of the fuel assemblies with the lower core plate (LCP) during reactor operation.

The upper support columns vertically position the UCP and are designed to take the uplifting hydraulic flow loads and fuel spring loads on the UCP. Plants that use the larger diameter columns have slots on the columns to allow coolant exiting from the fuel assemblies to enter the column and exit from the slotted holes. Mixing devices on some plants are provided on the UCP and also at the bottom of the upper support columns at the thermocouple locations. The guide tubes are bolted to the upper support plate and pinned at the UCP so they can be more easily removed if replacement is desired. The guide tubes are designed to guide the control rods in and out of the fuel assemblies to control power generation. The guide tubes are also slotted in their lower sections to allow coolant exiting from the core to flow into the upper plenum.

The upper instrumentation columns are bolted to the upper support plate. These columns support the thermocouple guide tubes that lead the thermocouples from the reactor head into the upper plenum to just above the UCP. In several plants, the thermocouples are combined with the Bottom Mounted Instrumentation (BMI).

The reactor core is positioned and supported by the lower internals and upper internals assembly. The individual fuel assemblies are positioned by fuel pins in the LCP and in the UCP. These pins

control the orientation of the core with respect to the lower internals and upper internals. The lower internals are aligned with the upper internals by the UCP alignment pins and secondarily by the head/vessel alignment pins. The lower internals are orientated to the vessel by the lower radial keys and by the head/vessel alignment pins. Therefore, the core is aligned with the vessel by a number of interfacing components.

The fuel assemblies are supported inside the lower internal assembly on top of the LCP. The LCP is elevated above the lower support forging by support columns and bolted to a ring support attached to the inside diameter (ID) of the core barrel. The support columns transmit vertical fuel assembly loads from the LCP to the much thicker lower support forging. The lower support forging is welded to and supported by the core barrel, which transmits vertical loads to the vessel through the core barrel flange. In the XL plants (14-foot core), the fuel assemblies are supported directly on the lower support forging.

The guidance and alignment of the lower assembly during insertion into the reactor vessel is provided by the vessel guide studs and the lower radial support keys and finally at the flange by the head/vessel alignment pins. The assembly is then supported by the core barrel flange that rests on the reactor vessel ledge. The hold-down spring is positioned on top of the flange and holds the lower assembly down, resisting the flow uploads. Horizontal loads placed on the lower internals assembly are reacted at the flange-to-vessel interface and at the lower radial support system.

When the coolant enters the reactor vessel, it impinges on the side of the core barrel and is directed downwards through the annulus formed by the gap between the outside diameter (OD) of the core barrel and the ID of the vessel. The flow then enters the plenum area between the bottom of the lower core barrel assembly and the vessel and is redirected upward through the core. After passing through the core, the coolant enters the upper internals region, then exits radially out through the core barrel/reactor vessel outlet nozzles. A small amount of flow is directed into the reactor vessel head area by the head cooling spray nozzles and into the former region (area between the baffle plates and the ID of the core barrel) for cooling of the baffle/former assembly. The perforations in the various components (such as, the lower support forging, the LCP, and diffuser plate) control and meter the flow through the core.

The following sections discuss the Westinghouse reactor internals components listed in Section 3 of the Issue Management Table (IMT) [C1].

C.1 Upper Internals Assembly

C.1.1 Upper Support Plate Assembly

The upper support plate assembly positions and supports the guide tubes and the upper support columns that, in turn, position and support the UCP. The upper support plate also positions and supports the thermocouple columns and guides. There are three models of upper support plate assemblies: 1) a deep beam, 2) a top hat, and 3) an inverted top hat.

The upper support plate consists of a perforated plate that is reinforced underneath by a stiffener ring and a deep beam structure. During reactor operation, the upper support plate is preloaded against the vessel head flange by the core hold-down spring. The hold-down spring rests on the core barrel flange, which prevents the lower internals as well as the upper internals from shifting due to flow forces. Because of the stiffener ring and the deep beam structure underneath the upper support plate, the upper support plate assembly is a stiff structure in the axial direction.

The upper support plate assembly is designed to sustain deadweight loads from the guide tubes, upper support columns, UCP, upper instrumentation columns, and the mixing devices. It is designed for mechanical loads from the fuel assembly hold-down spring forces and the core hold-down spring forces. It is designed for hydraulic loads, flow-induced vibration loads, thermal loads (normal operation and upset condition thermal transients), seismic (operational basis earthquake (OBE) and safe shutdown earthquake (SSE)), loss-of-coolant-accident (LOCA), and normal handling loads.

C.1.2 Upper Core Plate and Fuel Alignment Pins

The UCP positions the upper ends of the fuel assemblies and the lower ends of the control rod guide tubes, therefore serving as the transition member for the control rods in entry and retraction from the fuel assemblies. It also controls coolant flow when it exits from the fuel assemblies and serves as a boundary between the core and upper plenum.

The UCP is restrained from vertical movement by the upper support columns, which are attached to the upper support plate. The lateral movement is restrained by four alignment pins at each of the 4 major reactor axes. These pins are welded to the core barrel and interface with the core plate through core plate inserts, which are customized at manufacture. On the bottom side of the UCP, there are fuel pins (2 for each fuel assembly) for positioning and supporting the fuel assemblies. These pins in the XL model (14-foot core) are integral with the fuel assemblies.

The UCP assembly is designed to sustain deadweight loads from the UCP and the mixing devices. It is designed for mechanical loads from the fuel assembly hold-down spring forces. It is designed for hydraulic loads, flow-induced vibration loads, thermal loads (normal operation and upset condition thermal transients), gamma heating, seismic (OBE and SSE), LOCA, and normal handling loads.

C.1.3 Upper Support Column Assemblies

The upper support columns preload the fuel assembly and react to the fuel assembly forces, serve as separation members for the upper support plate and UCP in formation of the core outlet plenum, and position, guide, and support the thermocouples for core outlet water temperature measurement.

The upper support column is designed to sustain deadweight loads from the upper support columns, the UCP, and the mixing devices. It is designed for mechanical loads from the fuel

assembly hold-down spring forces. It is designed for hydraulic loads, flow-induced vibration loads, thermal loads (normal operation and upset condition thermal transients), seismic (OBE and SSE), LOCA, and normal handling loads.

C.1.4 Control Rod Guide Tube Assemblies (and Flow Downcomers)

The guide tubes (GTs) are bolted from the top of the upper support plate and are supported at their lower end to the UCP with spring-type pins. They provide a straight low-friction path for the control rods into or out of the fuel assemblies, provide sufficient protection for the control rods when they are withdrawn from the fuel elements to prevent damage due to parallel and lateral coolant flow, and provide a convenient and safe storage place for the control rod drive lines during refueling.

The GTs must be of sufficient strength to withstand both the dynamic and static loads imposed by the reactor coolant flow for both steady-state and transient operation. The GTs must also withstand loads imposed during accident conditions. In the event of a damaged drive line or stuck rod, the GTs are designed to be removable and replaceable without damage to reactor internals structure.

Each GT consists of two or three individual welded assemblies depending on the model. A removable insert at the top of the GT is held in place by flexures in 14x14 and 15x15 plants. The insert acts as a flow restrictor around the drive shaft to minimize bypass flow into the head plenum. The insert is designed to be removable to access, when necessary, the control rod drive mechanism drive shaft during refueling.

The GTs are designed to sustain axial loads for the control rod "scram" and stepping load, and to sustain deadweight loads from the guide tubes. They are designed for hydraulic crossflow loads exiting through the outlet nozzle, flow-induced vibration loads, pump-induced pressure loads, thermal loads (normal operation and upset condition thermal transients), seismic (OBE and SSE), LOCA, and normal handling loads.

C.1.5 Upper Instrumentation Conduit and Supports

The upper instrumentation columns provide a passageway and crossflow protection to the conduits that, in turn, house the thermocouples. The thermocouples are inserted into the top of the upper instrumentation columns and are routed down through the inside of various support columns. The ends of the thermocouples protrude below the upper support columns so that the temperature of the coolant exiting the fuel assemblies can be measured.

C.1.6 Mixing Devices

Mixing devices are used with thermocouples to enhance the temperature reading at the core outlet just above the UCP in 14x14 and 15x15 cores. (The mixing devices were not used in

plants using 17x17 and 16x16 cores that converted to the inverted upper support structure.) The mixing devices are cast cylinders with four vanes cast on the inside. They are located individually on the UCP or welded to the upper support columns at all thermocouple locations. They sustain the same loads as the upper support columns except when individually attached to the UCP.

C.2 Lower Internals Assembly

C.2.1 Lower Core Plate and Fuel Alignment Pins

The functions of the LCP are to position and support the core and provide a metered control of reactor coolant flow into each fuel assembly. The LCP is located near the bottom of the lower support assembly, inside the core barrel, and above the lower support forging. There are fuel pins attached to the core plate that position the fuel assemblies. The fuel assemblies are positioned over flow holes that control the amount of water entering each fuel assembly. The LCP is bolted at the periphery to a ring welded to the ID of the core barrel. The span of the plate is supported by lower support columns that are attached at their lower end to the lower support plate. At the center, a removable plate is provided for access to the vessel lower head region.

The LCP is designed to sustain deadweight loads from the core, the LCP, and the upper internals assembly. It is designed for mechanical loads from the fuel assembly spring forces and the control rod "scram" impact loads. It is designed for hydraulic loads, flow-induced vibration loads, thermal loads (normal operation and upset condition thermal transients, and gamma heating), seismic (OBE and SSE), LOCA, and normal handling loads.

The fuel alignment pins installed on the LCP engage the bottom nozzle of the fuel assembly. These pins provide the initial alignment of the fuel as the upper internals are lowered into place and react to the lateral loads from the fuel assembly at the bottom nozzle. The alignment pins installed on the UCP also provide a means of aligning, locating, and maintaining the position of the top nozzles of the fuel assemblies. For some plants, the fuel alignment pins are an integral part of the fuel assembly.

C.2.2 Lower Support Forging or Casting

A function of the lower support forging or casting is to provide support for the core by reacting to the LCP loads transmitted through the lower support columns. The plate must direct coolant flow from the lower head plenum to the core region. The lower support forging is attached to the lower end of the core barrel. The core rests directly on the LCP, which is supported by the lower support columns that are attached to and extend above the lower support forging. There is a large-diameter removable plate that provides access to the lower head plenum. The other through-holes direct flow from the lower head to the lower plenum (the area between the LCP and lower support forging) and permit instrumentation guide columns to pass through the support. On the outer periphery of the lower support forging are radial support keys welded into

machined pockets. Four-loop plants have six radial supports, and three- and two-loop plants have four radial supports. The BMI assembly and secondary core support system are attached to the underside of the support forging. For some plants, a diffuser plate is clamped in place by the support columns between the LCP and lower support forging. Some four-loop plants employ a cast lower support instead of a forging. The functions, loads, and supporting hardware are the same except for dimensions. For XL plants, the LCP, diffuser plate, and lower support columns were eliminated, and the fuel assemblies are supported directly by the lower support forging.

The lower support forging is designed to sustain deadweight loads from the core, the LCP, the lower supports, the diffuser plate, the BMIs, the secondary core support, and the upper internals assembly. It is designed for mechanical loads from the fuel assembly spring forces and the control rod "scram" impact loads. It is designed for hydraulic loads, flow-induced vibration loads, thermal loads (normal operation and upset condition thermal transients, and gamma heating), seismic (OBE and SSE), LOCA, and normal handling loads.

C.2.3 Lower Support Column Assemblies

The function of the lower support columns is to support the LCP and transmit the loads from the LCP to the lower support forging. Some lower support columns also serve as a guide for the neutron flux thimbles. The lower support columns separate the LCP and the lower support. The columns react to the core loads acting on the LCP and transmit these loads to the lower support. The columns are attached with threaded fasteners to the LCP and a threaded joint to the lower support.

The lower support columns are designed to sustain deadweight loads from the core, the LCP, the lower support columns, the diffuser plate, and the upper internals assembly. It is designed for mechanical loads from the fuel assembly spring forces and the control rod "scram" impact loads. It is designed for hydraulic loads, flow-induced vibration loads, thermal loads (normal operation and upset condition thermal transients, and gamma heating), seismic (OBE and SSE), LOCA, and normal handling loads.

C.2.4 Core Barrel

The primary function of the core barrel is to support the core. Lateral support for the core is provided at the upper and lower core plate locations and at intermediate positions during a seismic and LOCA event. During a seismic and LOCA event, the core may impact the baffle/former assembly that is supported by the core barrel. The core barrel also must provide a passageway for the reactor coolant flow. It directs the reactor coolant flow to the core and, after leaving the core, directs the flow to the outlet nozzles.

The core barrel outlet nozzles direct the flow radially outward through the reactor vessel outlet nozzles. The nozzles are located in the upper portion of the core barrel directly below the flange and are attached to the core barrel, each in line with a vessel outlet nozzle. A small amount of bypass leakage may occur in the gap between the core barrel outlet nozzle face and the vessel

outlet nozzle land. The nozzles extend radially from the core barrel to the ID of the vessel. The nozzles are classified as internal structures since they do not provide support for the core.

The core rests directly on the LCP that is ultimately supported by the core barrel. The LCP is attached at its periphery to the core barrel ID and supported by lower support columns that are attached to the lower support forging. The lower support forging is welded at its edge to the bottom end of the core barrel.

Four alignment pins located at 90-degree intervals are welded to the core barrel and engage the UCP. These pins restrain the lateral motion of the UCP. The baffle/former assembly is bolted to the core barrel and forms an outer envelope for the core. Attached to the core barrel or to the core barrel flange are the baffle/former assembly, outlet nozzles, neutron panel assemblies or thermal shield, alignment pins, the LCP, the lower support forging, vessel alignment pins, specimen plugs, and head cooling spray nozzles.

The core barrel is designed to sustain deadweight loads from the core, the LCP, the lower supports, the baffle assembly, the core barrel, the attached internals structures, and the upper internals assembly. It is designed for mechanical loads from the fuel assembly and hold-down spring forces and control rod "scram" impact loads. It is designed for hydraulic loads, flow-induced vibration loads, thermal loads (normal operation and upset condition thermal transients, and gamma heating), seismic (OBE and SSE), LOCA, and normal handling loads.

C.2.5 Radial Support Keys

The radial keys restrain large transverse motions of the core barrel but, at the same time, allow unrestricted radial and axial thermal expansions. The lower core barrel is restrained laterally and torsionally by these uniformly spaced keys. The radial keys, along with the matching clevis inserts, are designed to limit the tangential motion between the lower end of the core barrel and the vessel. The keys engage the keyways of the inserts in the axial direction such that the core barrel is provided with a support at the farthest extremity, that is, fixed at the top and guided at the bottom. With the radial key and inserts, the radial and axial expansions of the core barrel are accommodated but circumferential movement (that is, rotation) of the core barrel is restricted. The radial keys are attached to the core barrel at the lower support forging level. The inserts are attached to a clevis that is welded to the vessel. The contact surfaces on the radial keys are surface-hardened to increase their wear resistance.

The radial keys and the clevis inserts are designed to sustain vibratory loads, steady-state interference loads from the core, and vertical frictional force due to differential thermal growth relative to that of the core barrel. They are designed for seismic (OBE and SSE) and LOCA loads.

Failure of one of the radial support keys would place more stress on the remaining support keys to maintain lateral and torsional restraint on the core barrel. The barrel is not expected to become unstable with the failure of one radial support key. Failure of a radial support key bolt or lock

key is not expected to result in the failure of the radial support key, although the economic consequence could be significant.

C.2.6 Baffle and Former Assembly

The baffle and former assembly is made up of vertical plates called baffles and horizontal support plates called formers. The baffle plates are bolted to the formers by the baffle/former bolts, and the formers are attached to the core barrel inside surface by the barrel/former bolts.

The baffle/former assembly forms the interface between the core and the core barrel. The baffles provide a barrier between the core and the former region so that a high concentration of flow in the core region can be maintained. A secondary benefit is to reduce the neutron flux on the vessel. The baffle and former assembly also provide lateral support for the core during a seismic or LOCA event.

The baffle and former assembly extends the full length of the core and follows the peripheral contour of the core. This restricts most of the coolant flow to the core area by keeping the flow out of the former region. The formers, which are bolted on their OD to the core barrel ID, position and provide structural support for the baffle plates. Some of the baffle plates are also bolted to each other at selected corners by edge bolts or brackets.

The baffle/former assembly is designed to sustain deadweight loads from the baffle/former assembly. It is designed for mechanical loads from the bolt pre-loads. It is designed for hydraulic loads, flow-induced vibration loads, thermal loads (normal operation and upset condition thermal transients, and gamma heating), seismic (OBE and SSE), LOCA, and normal handling loads.

C.2.7 Neutron Panels/Thermal Shield

Additional neutron shielding of the reactor vessel is provided in the active core region by neutron panels or thermal shields that are attached to the outside of the core barrel. Neutron panels are attached to the OD of the core barrel at strategically located positions to reduce the fluence on the reactor vessel welds. The thermal shield design provides shielding for the complete 360-degree circumferential sector. It is fastened with bolts and dowels below the outlet nozzles and also near the lower portion of the core barrel with flexures.

C.2.8 Irradiation Specimen Guides

Specimen guides contain surveillance specimens for monitoring the effects of irradiation on the reactor vessel during the life of the plant. These are attached to the neutron panels or thermal shields. At specific intervals during the design life of the reactor, a specimen container will be removed from the reactor vessel and the material samples will be tested to determine the irradiation effects on the reactor vessel.

Failure of a specimen guide would necessitate replacement, although the economic consequence is not significant.

C.2.9 Secondary Core Support Assembly

The functions of the secondary core support is to absorb a portion of the energy generated by the downward displacement of the core barrel subassembly during a postulated failure and limit the force imposed on the vessel, to transmit and distribute the vertical load of the core to the reactor vessel, to limit the displacement to prevent withdrawal of the control rods from the core, and to limit the displacement to prevent loss of alignment of the core with the upper core support to allow the control rods to scram.

The secondary core support is provided in the plenum area between the bottom of the core barrel subassembly and the bottom of the reactor vessel. To prevent the control rods from being withdrawn and to limit the load to the vessel, the system is designed to absorb this kinetic energy from the vertical displacement within a limited distance. A curved bearing plate attached to the bottom of the energy absorber distributes these loads to the vessel. The support system comprises four support columns bolted at one end to the underside of the lower support forging and bolted at the bottom to an energy absorbing device. Each energy absorber is made up of three concentric cylinders, one of which is a custom-machined cylinder designed to absorb the kinetic energy of the lower internals assembly and the core. The four energy absorbers are seated in a base plate that is contoured to the approximate shape of the lower head.

C.2.10 Bottom-Mounted In-core Instrumentation Column Assemblies

The function of these columns is to provide an insertion path for the flux thimbles into the core from the bottom of the vessel and to protect the flux thimbles during the operation of the reactor. There are two types of BMI columns. The cruciform columns extend through the flow holes of the lower support forging and attach to the bottom of the LCP. The standard guide columns line up with the lower support columns and are bolted to the bottom side of the lower support. This provides an uninterrupted, protected path for flux thimbles entering the reactor core.

C.2.11 Head Cooling Spray Nozzles

Head cooling spray nozzles are used to adjust the upper plenum coolant temperature by allowing bypass flow at the vessel inlet temperature from the vessel/core barrel downcomer region to flow directly into the upper head plenum. Several different designs evolved, so the exact configuration would depend on the production date.

The latest design employs tubes welded to the core barrel flange. These tubes extend through openings in the upper support plate at a diameter outside of the hold-down spring. The tubes are fitted with an adjustable orifice. The orifice is threaded and crimped in place. Other designs were

similar, except they did not have an orifice attachment and directed a smaller portion of the bypass flow (inlet) to the upper head, leading to higher bulk average upper head temperatures.

C.2.12 Diffuser Plate

To enhance flow uniformity entering the LCP, some plants employ an additional orifice plate called a diffuser plate. This plate is clamped in place by the lower support columns between the LCP and lower support plate.

C.2.13 Flux Thimbles

The flux thimble is a long, slender stainless steel sealed tube that passes through the vessel penetration, through the lower internals assembly, and finally extends to the top of the fuel assembly. The flux thimble tube provides a path for the neutron flux detector into the core and is subjected to reactor coolant pressure and temperature on the outside surface and to atmospheric conditions on the inside.

The flux thimbles remain stationary during reactor operation, with the bullet end of the thimbles positioned slightly above the top of the active fuel. For refueling, the thimbles are retracted to a point where the bullet tip is below the LCP. For the removal of the lower internals assembly, the flux thimbles are pulled out further until the bullet tip is outside of the reactor vessel.

C.3 Interfacing Components

The general requirements of the interfacing components are to orient adjacent components with respect to each other and/or provide support for an adjacent component. These components are the lower internals assembly, the upper internals assembly, the fuel and driveline, or the reactor vessel. The UCP alignment pins position the UCP with respect to the lower internals assembly and provide lateral support to the lower end of the upper internals assembly. The hold-down spring supports the upper internals assembly and holds the lower internals assembly down. The head and vessel alignment pins align the lower internals assembly and the upper internals assembly with the vessel. The radial support inserts provide a support surface for the radial support keys.

C.3.1 Hold-Down Spring

The hold-down spring provides a preload to limit the axial motion of the upper and lower internals assemblies and to prevent the lift-off of the core barrel flange from the vessel ledge. The spring preload also reduces the lateral motion of the upper support plate flange and the core barrel flange. The hold-down spring, which is a circumferential spring with an essentially rectangular cross-section, is located between the flanges of the upper support plate and the core barrel. The hold-down spring is preloaded by a compressive force when the reactor vessel head is clamped in place with the reactor vessel closure studs and nuts. Therefore, the hold-down spring

is an interfacing component between the upper internals assembly and the lower internals assembly. The hold-down spring is normally left with the lower internals during refueling.

The hold-down spring is designed to maintain a spring preload force and to react to loads from the relative movement of the flanges, flow loads, various thermal transients, and OBE seismic loads.

C.3.2 Upper Core Plate Alignment Pin

The UCP alignment pins locate the UCP laterally with respect to the lower internals assembly. The pins must laterally support the UCP so that the plate is free to expand radially and move axially during differential thermal expansions between the upper internals and the core barrel.

The UCP alignment pins are the interfacing components between the UCP and the core barrel. The UCP alignment pins are shrunk-fit and welded into the core barrel and the core barrel bearing pad. The gap sizes between the alignment pins and the matching inserts are customized.

The UCP alignment pins are designed to sustain vibratory loads, steady interference loads, and vertical and radial friction loads due to differential thermal growth between the core barrel and the UCP, dynamic insertion loads, seismic loads (OBE and SSE), and LOCA loads.

C.3.3 Head and Vessel Alignment Pins

The head and vessel alignment pins align the upper and lower internals assemblies with respect to the vessel. The head vessel alignment pins are located at the outside periphery of the core barrel flange at the four major axes. A portion of the pin extends below the core barrel flange and engages pockets in the reactor vessel to provide alignment of the lower internals assembly with respect to the vessel.

Similarly, a portion of the pin extends above the flange and aligns the upper internals assembly with respect to the vessel. This portion of the pin engages pockets in the reactor vessel head, therefore establishing an alignment of lower internals, reactor vessel, upper internals, and reactor vessel head. Minimal clearance is maintained between the pins and the engagement pockets to ensure functional alignment and to allow ease of assembly. The clearances are designed to prevent thermal loads in the pins during temperature excursions and to reduce the stress in the pins during horizontal loading of the upper internals.

C.3.4 Clevis Inserts

The discussion of this component is summarized in Section 3.2.5.

C.4 Appendix C Reference

C1. *Pressurized Water Reactor Issue Management Tables, PWR-IMT Consequence of Failure (MRP-156)*, Palo Alto, CA, 2005, TP-1012110.

D COMPONENT FUNCTIONAL DESCRIPTIONS FOR COMBUSTION ENGINEERING REACTOR INTERNALS

Components and Functions Overview of the Combustion Engineering Pressurized Water Reactor (PWR) Internals

The Combustion Engineering (CE) PWR internals assembly is part of the reactor coolant system and is located inside the reactor pressure vessel. The reactor internals are long-lived passive structural components. The intended functions are to support core cooling, enable control rod insertion, and maintain the integrity of the fuel. Internals components are classified as either core support structures or internals structures. The major components are the upper internals assembly, the control element assembly (CEA) shrouds, the core support barrel (CSB) assembly, the core shroud assembly, the lower internals assembly, the control element assemblies and the in-core instrumentation (ICI) system.

The upper internals assembly is located above the reactor core within the CSB and is removed during refueling as a single component. The upper end of the assembly is flanged and rests upon the CSB. The upper internals assembly functions are to provide alignment and support of the fuel assemblies, to maintain CEA shroud spacing, to prevent movement of the fuel assemblies in the case of a severe accident condition, and to protect the control rods from crossflow effects in the upper plenum. The CEA shroud assemblies consist of the CEA shrouds, the CEA shroud bolts, and the CEA shroud extension shaft guides. The CEA shroud assemblies are located within the upper internals assembly above the fuel alignment plate (FAP), and extend to an elevation above the upper guide structure support plate.

The CSB assembly consists of the CSB, the CSB upper flange, the CSB alignment keys, the CSB snubbers, and, for a single plant, the thermal shield. The core shroud assembly is located within the CSB and below the upper internals assembly. The core shroud assembly is attached to the core support plate (CSP) by tie rods, bolts, or welds. The core shroud assembly provides a boundary for the coolant flow and limits the amount of coolant bypass flow. The core shroud assembly also reduces the lateral motion of the fuel assemblies.

The lower internals assembly positions and provides axial support for the core. The fuel assemblies are supported by the CSP, which is supported by core support columns resting on lower support structure beam assemblies. The lower internals assembly is welded to the lower flange of the CSB.

The ICI system consists of the ICI assemblies and ICI support assemblies. A fixed number of ICI assemblies are inserted into selected fuel assemblies. The ICI assemblies are installed in the reactor core at selected locations to obtain core neutron flux and coolant temperature information during reactor operation. The ICI support assembly supports and aligns the ICI assemblies within the upper internals assembly.

The following sections discuss the Combustion Engineering reactor internals components listed in Section 9 of the Issue Management Table (IMT) [D1].

D.1 Upper Internals Assembly

The upper internals assembly (or upper guide structure) consists of the upper guide structure support plate, the fuel assembly alignment plate, the FAP guide lugs and inserts, and the hold-down ring (HDR) or expansion compensating ring. The upper guide structure support plate consists of a top plate, a cylindrical skirt, and a grid network. The skirt and grid network are welded to the underside of the top plate. There are slots in the top plate for alignment with the core support barrel and reactor vessel. The fuel assembly alignment plate is attached to the bottom of the upper internals assembly and has slots that align it with the core shroud assembly. The function of the FAP is to align the upper ends of the fuel assemblies and to support and align the lower ends of the CEA shrouds that are bolted to the FAP. The fuel assemblies are aligned with precision-machined holes in the FAP. The FAP also has slots on its outer edge that engage the FAP guide lugs protruding up from the core shroud to limit lateral motion of the upper internals assembly during operation.

The FAP guide lugs are attached to the top of the core shroud and act as mating keyways. In one plant, the FAP guide lugs are bolted to the inside diameter of the CSB. In others, the FAP guide lugs are bolted to the core shroud, or they are welded to the core shroud. This interface establishes the horizontal location of the FAP and hence the upper guide structure, CEA shrouds, and CEAs with respect to the core shroud and hence the lower internals assembly, core, and CSB. This alignment of the fuel assemblies with the CEAs ensures the ability to withdraw and insert the CEAs in a controlled manner. In addition, for those plants having ICI entering through the top of the reactor, this alignment allows the insertion and removal of the ICI assemblies.

The HDR or expansion compensating ring provides structural support to the fuel and CEAs by preloading the upper internals assembly and the CSB. The HDR acts as a shim and is set between the reactor vessel closure head flange and the upper internals assembly to resist axial upward movement. This arrangement accommodates differential axial thermal expansion between the CSB flange, the upper internals assembly flange, the reactor vessel mating surface and the reactor vessel closure head flange recess. In three plants, the HDR is located between the underside of the upper internals assembly flange and the top of the CSB flange.

D.2 Lower Support Structure

The lower support structure positions consists of the CSP, the fuel alignment pins, the core support columns, and the lower support structure beam assemblies. The CSP positions and supports the reactor core and provides control of reactor coolant flow into each fuel assembly. The fuel alignment pins protrude from the CSP and provide guidance and limit lateral movement of the individual fuel assemblies. Three plants do not have a CSP, so the fuel alignment pins are attached directly to the core support columns.

The core support columns transfer the loads from the CSP to the lower support structure beam assemblies. The lower support structure beam assemblies then transfer the loads to the CSB. The top of the core support column is attached to the CSP while the bottom of the column is welded to the lower support structure beam assembly. There are no core support columns in three CE-designed plants. The lower support structure beam assemblies are welded to the bottom of the core support columns, which are attached to the CSP. In one plant, bolts are used to attach the CSP to the core support columns.

D.3 Control Element Assemblies (CEA)-Shroud Assemblies

The CEA shroud assemblies consist of the CEA shrouds, the CEA shroud bolts, and the CEA shroud extension shaft guides. The CEA shroud assemblies are located within the upper internals assembly above the FAP, and extend to an elevation above the upper guide structure support plate. The CEA shroud assemblies provide support, alignment and spacing of the CEAs and ICIs, and they provide flow passages for coolant passing through the FAP while protecting the CEAs from crossflow.

The CEA shrouds consist of the shroud tubes, shroud bolts, and the shroud extension shaft guides. The CEA shrouds guide the control rods into and out of the core. The bottom part of the CEA shrouds is bolted at the lower end to the fuel assembly alignment plate. The CEA shroud extension shaft guides provide lateral support and alignment of the CEAs above the upper guide structure support plate by aligning the CEAs with the CEA shrouds. Additionally, the CEA extension shaft guides protect the CEAs from crossflow, albeit very small, in the upper reactor vessel head. The control element drive mechanisms (CEDMs) are positioned on the reactor vessel closure head and are coupled to the CEAs by the CEA extension shafts.

D.4 Core Support Barrel Assembly

The CSB assembly consists of the CSB, the CSB upper flange, the CSB alignment keys, the CSB snubbers, and at one plant, the thermal shield. The CSB is cylindrically shaped and contains the core and other internals. The CSB absorbs the static loads from the fuel assemblies and other components, and the dynamic loads from normal operating hydraulic flow, seismic disturbances, and loss-of-coolant-accident (LOCA) events.

The CSB upper flange is a thick ring flange that supports and suspends the CSB from a ledge on the reactor vessel. The CSB supports the lower internals assembly and the CSP upon which the fuel assemblies rest. The CSP transmits the weight of the core to the CSB by means of vertical columns, an annular skirt and a beam structure. The reactor vessel, closure head and upper internals assembly flanges are slotted in locations corresponding to the CSB alignment keys to provide proper alignment between these components in the vessel flange region.

The CSB alignment keys align the CSB assembly relative to the reactor vessel and reactor vessel head. The alignment keys also align the upper internals assembly with the CSB assembly. The alignment keys are press-fitted into the top flange of the CSB. The CSB snubber assemblies limit lateral movement of the CSB during postulated accidents such as seismic and LOCA events. The CSB is supported at the upper end, which allows coolant flow-induced vibrations into the structure. In order to prevent excessive vibration, snubber assemblies are installed on the outside of the bottom of the CSB to limit lateral motion. The snubbers consist of equally spaced blocks installed around the circumference of the CSB, while the core stabilizing lugs are attached to the reactor vessel. The lugs are attached to the reactor vessel. The snubbers and core stabilizing lugs comprise a "tongue-and-groove" assembly, where the snubbers are the grooves and the core stabilizing lugs are the tongues.

Only one plant has an installed thermal shield. It is a cylindrical structure that reduces the neutron flux and radiation heating in the reactor vessel wall. At the upper end, it is supported by pins that rest upon equally spaced lugs on the outer periphery of the CSB. There are shims bolted to the underside of the lugs that reduce the gap between the bottom of the lug and the thermal shield. The lower end of the thermal shield is positioned radially by pins that pass through the shield and butt against the CSB. These positioning pins, or thermal shield bolts, hold the thermal shield tightly against the CSB.

D.5 Core Shroud Assembly

The core shroud assembly consists of the core shroud, the ribs (formers), and tie rods. The core shroud assembly is attached to the CSP by tie rods, bolts, or welds. The core shroud provides an envelope for the core and limits the amount of coolant bypass flow. The shroud consists of a vertical assembly of plates designed to channel the coolant through the core. Circumferential rings (ribs or former plates) and a top and bottom end plate provide lateral support. The vertical plates are bolted to the formers at two plants. The plates are welded together and to circumferential rings or ribs in the case of the other CE plants.

A small gap is provided between the core shroud outer perimeter and the CSB to provide upward coolant flow in the annulus, thereby minimizing thermal stresses in the core shroud and CSB. Some plants have flow passages in the former plates to facilitate flow up the annulus. Alignment lugs (FAP guide lugs) protrude vertically from the top of the core shroud and engage in corresponding slots in the upper internals assembly fuel assembly alignment plate to ensure proper alignment between the upper internals assembly, core shroud, and lower internals assembly.

Welds, bolts, or tie rods hold the core shroud segments together and attach these segments to the CSP. Additionally, at two plants the upper and lower sections of the core shroud are aligned with radial pins. At two other plants the former plates are bolted to the CSB.

D.6 In-Core Instrumentation (ICI) System

The ICI system consists of the ICI assemblies and an ICI support assembly. There are two basic CE designs: top-entry and bottom-entry ICIs.

For the plants with top-entry ICIs, the ICI assemblies consist of an ICI guide tube and an ICI thimble. The ICI is contained within the ICI thimble, which in turn is contained within the ICI guide tube. The ICI assemblies are inserted into the core through a number of instrumentation nozzles in the reactor vessel head. Each assembly is guided into position in an empty fuel tube in the center of the fuel assembly via a fixed guide tube.

For the plants with top-entry ICIs, the ICI support assembly consists of an ICI support plate that supports the ICI guide tubes and thimbles. The ICI support assembly has two functions:

- To provide full-length support and guidance of the in-core instruments from the pressure boundary flange to the selected fuel bundle location
- To protect the in-core detector assemblies from flow effects within the reactor vessel

The ICI support plate is perforated to fit over the CEA extension shaft guides. The ICI support plate is supported in the upper guide structure by pins that rest on the upper guide structure support plate. The upper guide structure CEA extension shaft guides extend through holes in the ICI support plate. The ICI thimbles extend downward through the ICI guide tubes in the upper guide structure into fuel bundles. Above the ICI support plate, the ICI guide tubes bend and are gathered to form stalks called guide tube clusters that extend into the reactor vessel closure head instrumentation nozzles.

The ICI guide tubes are stationary components in the fuel assemblies and the reactor vessel internals through which flexible ICI thimbles containing the ICI are inserted into and removed from the core. The ICI guide tubes are supported next to a certain number of the CEA shrouds. Each ICI guide tube is attached to a CEA shroud with a strap that is welded on one end to the guide tube and on the other end to the CEA shroud. The ICI guide tubes interface with the FAP.

The ICI thimbles are flexible tubes that contain the ICI tubes. These tubes are inserted into the core through the ICI guide tubes. The ICI thimbles fill the space between the ICI guide tubes and the in-core instruments in order to prevent the ICI from vibrating against the inside of the guide tubes as a result of the coolant flow.

Three CE-designed plants are equipped with bottom-entry ICIs. For this design, the ICI assemblies consist of an ICI nozzle (comparable to the ICI guide tube for the top-entry plants). The ICI nozzle contains the ICI. There are no ICI thimbles in these plants. The ICI support

assembly is welded to the lower internals assembly (lower support structure). The ICI support assembly consists of the ICI nozzle support plate and the ICI nozzles. The bottom-entry ICI support assembly has two functions:

- To provide guidance and support to the ICI within the lower internals assembly
- To protect the in-core detector assemblies from flow effects within the reactor vessel

The ICI nozzles contain the ICI and penetrate through holes in the ICI nozzle support plate into the lower internals assembly. The ICI nozzles terminate at the grid located at the bottom of the fuel assemblies. The ICI then enters a guide tube located within the center of the fuel assemblies. There are no ICI thimbles in these plants.

D.7 Appendix D Reference

D1. *Pressurized Water Reactor Issue Management Tables, PWR-IMT Consequence of Failure (MRP-156)*, Palo Alto, CA, 2005, TP-1012110.

Export Control Restrictions

Access to and use of EPRI Intellectual Property is granted with the specific understanding and requirement that responsibility for ensuring full compliance with all applicable U.S. and foreign export laws and regulations is being undertaken by you and your company. This includes an obligation to ensure that any individual receiving access hereunder who is not a U.S. citizen or permanent U.S. resident is permitted access under applicable U.S. and foreign export laws and regulations. In the event you are uncertain whether you or your company may lawfully obtain access to this EPRI Intellectual Property, you acknowledge that it is your obligation to consult with your company's legal counsel to determine whether this access is lawful. Although EPRI may make available on a case-by-case basis an informal assessment of the applicable U.S. export classification for specific EPRI Intellectual Property, you and your company acknowledge that this assessment is solely for informational purposes and not for reliance purposes. You and your company acknowledge that it is still the obligation of you and your company to make your own assessment of the applicable U.S. export classification and ensure compliance accordingly. You and your company understand and acknowledge your obligations to make a prompt report to EPRI and the appropriate authorities regarding any access to or use of EPRI Intellectual Property hereunder that may be in violation of applicable U.S. or foreign export laws or regulations.

The Electric Power Research Institute (EPRI)

The Electric Power Research Institute (EPRI), with major locations in Palo Alto, California, and Charlotte, North Carolina, was established in 1973 as an independent, nonprofit center for public interest energy and environmental research. EPRI brings together members, participants, the Institute's scientists and engineers, and other leading experts to work collaboratively on solutions to the challenges of electric power. These solutions span nearly every area of electricity generation, delivery, and use, including health, safety, and environment. EPRI's members represent over 90% of the electricity generated in the United States. International participation represents nearly 15% of EPRI's total research, development, and demonstration program.

Together...Shaping the Future of Electricity

Program:

Nuclear Power

© 2006 Electric Power Research Institute (EPRI), Inc. All rights reserved. Electric Power Research Institute and EPRI are registered service marks of the Electric Power Research Institute, Inc.

Printed on recycled paper in the United States of America

1013234

ELECTRIC POWER RESEARCH INSTITUTE

3420 Hillview Avenue, Palo Alto, California 94304-1338 • PO Box 10412, Palo Alto, California 94303-0813 USA

800.313.3774 • 650.855.2121 • askepri@epri.com • www.epri.com