Westinghouse Non-Proprietary Class 3

WCAP-17096-NP
Revision 2

December 2009

Reactor Internals Acceptance Criteria Methodology and Data Requirements

Westinghouse

WCAP-17096-NP
Revision 2

Reactor Internals Acceptance Criteria Methodology and Data Requirements

Randy Lott*
Reactor Internals Design and Analysis

Steve Fyfitch
Materials and Structural Analysis (AREVA NP Inc.)

December 2009

Reviewer: Joshua McKinley*
Reactor Internals Design and Analysis

Approved: Brian Gaia*, Manager
Reactor Internals Design and Analysis

Work Performed under PA-MSC-0473

*Electronically approved records are authenticated in the electronic document management system.

Westinghouse Electric Company LLC
P.O. Box 355
Pittsburgh, PA 15230-0355

LEGAL NOTICE

COPYRIGHT NOTICE

DISTRIBUTION NOTICE

PWR Owners Group
Member Participation* for PA-MSC-0473

Utility Member	Plant Site(s)	Participant	
		Yes	No
AmerenUE	Callaway (W)	X	
American Electric Power	D.C. Cook 1&2 (W)	X	
Arizona Public Service	Palo Verde Unit 1, 2, & 3 (CE)	X	
Constellation Energy Group	Calvert Cliffs 1 & 2 (CE)	X	
Constellation Energy Group	Ginna (W)	X	
Dominion Connecticut	Millstone 2 (CE)	X	
Dominion Connecticut	Millstone 3 (W)	X	
Dominion Kewaunee	Kewaunee (W)	X	
Dominion VA	North Anna 1 & 2, Surry 1 & 2 (W)	X	
Duke Energy	Catawba 1 & 2, McGuire 1 & 2 (W)	X	
	Oconee 1, 2, 3 (B&W)	X	
Entergy	Palisades (CE)	X	
Entergy Nuclear Northeast	Indian Point 2 & 3 (W)	X	
Entergy Operations South	Arkansas 2, Waterford 3 (CE)	X	
	Arkansas 1 (B&W)	X	
Exelon Generation Co. LLC	Braidwood 1 & 2, Byron 1 & 2 (W)	X	
	TMI 1 (B&W)	X	
FirstEnergy Nuclear Operating Co	Beaver Valley 1 & 2 (W)	X	
	Davis-Besse (B&W)	X	
Florida Power & Light Group	St. Lucie 1 & 2 (CE)	X	
	Turkey Point 3 & 4, Seabrook (W)	X	
	Pt. Beach 1&2 (W)	X	
Luminant Power	Comanche Peak 1 & 2 (W)	X	
Xcel Energy	Prairie Island 1&2	X	
Omaha Public Power District	Fort Calhoun (CE)	X	
Pacific Gas & Electric	Diablo Canyon 1 & 2 (W)	X	
Progress Energy	Robinson 2, Shearon Harris (W)	X	
	Crystal River 3 (B&W)	X	
PSEG - Nuclear	Salem 1 & 2 (W)	X	
Southern California Edison	SONGS 2 & 3 (CE)	X	
South Carolina Electric & Gas	V.C. Summer (W)	X	
So. Texas Project Nuclear Operating Co.	South Texas Project 1 & 2 (W)	X	
Southern Nuclear Operating Co.	Farley 1 & 2, Vogtle 1 & 2 (W)	X	
Tennessee Valley Authority	Sequoyah 1 & 2, Watts Bar (W)	X	
Wolf Creek Nuclear Operating Co.	Wolf Creek (W)	X	

* **Project participants as of the date the final deliverable was completed. On occasion, additional members will join a project. Please contact the PWR Owners Group Program Management Office to verify participation before sending this document to participants not listed above.**

PWR Owners Group
International Member Participation* for PA-MSC-0473

Utility Member	Plant Site(s)	Participant	
		Yes	No
British Energy	Sizewell B	X	
Electrabel (Belgian Utilities)	Doel 1, 2 & 4, Tihange 1 & 3	X	
Hokkaido	Tomari 1 & 2 (MHI)	X	
Japan Atomic Power Company	Tsuruga 2 (MHI)	X	
Kansai Electric Co., LTD	Mihama 1, 2 & 3, Ohi 1, 2, 3 & 4, Takahama 1, 2, 3 &4 (W & MHI)	X	
Korea Hydro & Nuclear Power Corp.	Kori 1, 2, 3 & 4 Yonggwang 1 & 2 (W)	X	
Korea Hydro & Nuclear Power Corp.	Yonggwang 3, 4, 5 & 6 Ulchin 3, 4 , 5 & 6(CE)	X	
Kyushu	Genkai 1, 2, 3 & 4, Sendai 1 & 2 (MHI)	X	
Nuklearna Electrarna KRSKO	Krsko (W)	X	
Nordostschweizerische Kraftwerke AG (NOK)	Beznau 1 & 2 (W)	X	
Ringhals AB	Ringhals 2, 3 & 4 (W)	X	
Shikoku	Ikata 1, 2 & 3 (MHI)	X	
Spanish Utilities	Asco 1 & 2, Vandellos 2, Almaraz 1 & 2 (W)	X	
Taiwan Power Co.	Maanshan 1 & 2 (W)	X	
Electricite de France	54 Units	X	

* **This is a list of participants in this project as of the date the final deliverable was completed. On occasion, additional members will join a project. Please contact the PWR Owners Group Program Management Office to verify participation before sending documents to participants not listed above.**

TABLE OF CONTENTS

LIST OF TABLES

LIST OF FIGURES

None.

LIST OF ACRONYMS AND ABBREVIATIONS

ANO	Arkansas Nuclear One
ASME	American Society of Mechanical Engineers
B&W	Babcock & Wilcox
B&WOG	Babcock & Wilcox Owners Group
BB	baffle-to-baffle
BF	baffle-to-former
BMI	bottom-mounted instrumentation
BWR	boiling water reactor
BWRVIP	boiling water reactor vessel internals program
CASS	cast austenitic stainless steel
CE	Combustion Engineering
CEA	control element assembly
CF	core barrel-to-former
CGR	crack growth rate
CLB	current licensing basis
COD	crack opening displacement
CR	Crystal River
CRGT	control rod guide tube
CSS	core support shield
CW	cold worked
DB	Davis-Besse
DNB	departure from nucleate boiling
EFPY	effective full-power year
EPRI	Electric Power Research Institute
ET	eddy-current test
FD	flow distributor
FEA	finite element analysis
FEM	finite element model
FMEA	failure modes and effects analysis
HAZ	heat-affected zone
HWC	hydrogen water chemistry
I&E	inspection and evaluation
IASCC	irradiation-assisted stress corrosion cracking
IC	irradiation creep
ID	inside diameter
IE	irradiation embrittlement
IGSCC	intergranular stress corrosion cracking
IMI	incore monitoring instrumentation
ISI	in-service inspection
ISR	irradiation-induced stress relaxation
JCO	justification for continued operation
LCB	lower core barrel
LEFM	linear-elastic fracture mechanics
LOCA	loss-of-coolant accident

LIST OF ACRONYMS AND ABBREVIATIONS (cont.)

LTS	lower thermal shield
MRP	Materials Reliability Program
NDE	nondestructive examination
NRC	U.S. Nuclear Regulatory Commission
NSSS	nuclear steam supply system
OD	outside diameter
OE	operating experience
ONS	Oconee Nuclear Station
PWR	pressurized water reactor
PWROG	Pressurized Water Reactor Owners Group
PWSCC	primary water stress corrosion cracking
RICT	Reactor Internals Core Team
RIFG	Reactor Internals Focus Group
SCC	stress corrosion cracking
SCF	stress concentration factor
SSE	safe-shutdown earthquake
SSHT	surveillance specimen holder tube
TAC	Technical Assignment Control
TBD	to be determined
TE	thermal embrittlement
TLAA	time-limited aging analysis
TMI	Three Mile Island
UCB	upper core barrel
UT	ultrasonic test
UTS	upper thermal shield
WOG	Westinghouse Owners Group

Trademark Statement:

WINCISE is a trademark of Westinghouse Electric Company LLC.

ACKNOWLEDGEMENTS

The authors would like to thank Jim Cirilli, Mike McDevitt, and the members of the PWROG Materials Subcommittee for their review and comments on Revision 0 of this WCAP. The authors would also like to acknowledge the contributions of their colleagues at AREVA-NP and Westinghouse in developing this document.

1 OBJECTIVE

The objective of this project is to identify consistent, industry-wide analytical methodologies and data requirements for developing:

1. **Acceptance Criteria** for the Primary and Expansion Components identified in the Materials Reliability Program (MRP) Reactor Internals Inspection and Evaluation (I&E) Guidelines (MRP-227 Rev. 0)

2. **Evaluation Procedures** for utilities to assess potential safety and functional impacts of degradation in components with observed relevant conditions

These criteria and procedures must be established and generally accepted across the industry prior to the implementation of the I&E Guidelines. This effort supports the I&E recommendations of MRP-227. It is anticipated that the methodologies and data requirements defined in this effort will be reviewed by the U.S. Nuclear Regulatory Commission (NRC) in the course of the MRP-227 Safety Evaluation.

The current project is only Phase I of the efforts required to develop acceptance criteria. The potential for generic efforts for each component is also evaluated in this program, and where practicable, work scope for a follow-on program to address the generic analyses of these components is identified. Because the component lists are specific to the original nuclear steam supply system (NSSS) suppliers, it is anticipated that any effort to develop acceptance criteria for specific components will be submitted to the Pressurized Water Reactor Owners Group (PWROG) for cafeteria funding.

For each of the Primary and Expansion Components listed in MRP-227, this report outlines:

- Type of analyses required
- Required evaluation procedures
- Data required to support analysis
- Logic chart illustrating evaluation path and potential disposition options
- Component items (Primary and Expansion) that can be addressed on a generic basis

Note that letter OG-09-290[1] issued Revision 0 of this document for review and comment by the PWROG Materials Subcommittee on July 27, 2009. Comments from this review have been incorporated in the current version.

[1] Letter OG-09-290, "Transmittal of Draft Report WCAP-17096, Rev 0 "Reactor Internals Acceptance Criteria Methodology and Data Requirements", PAMSC-0473," July 27, 2009.

2 BACKGROUND

The MRP Reactor Internals Focus Group (MRP RIFG) issued a formal request for the PWROG to sponsor a project to develop "generic" acceptance criteria for the proposed MRP-227 I&E Guidelines [1] on March 19, 2008 (Reference MRP Letter 2008-026). The MRP RIFG submitted the I&E Guidelines to the NRC for a formal Safety Evaluation in December 2008. At that time, the Guidelines would be submitted under NEI 03-08 Procedures. The schedule of this project was adopted to match the requirements for NRC review.

The I&E Guidelines in [1] for pressurized water reactor (PWR) internals are based on a thorough screening of both potential degradation and operating experience [2, 3, 4]. They are designed to target inspections of locations where aging degradation can potentially impair component function. Functionality analyses associated with the original screening evaluations have identified the potential operational concerns, and inspection methodologies have been identified for each component [5, 6]. However, the current I&E Guidelines do not provide detailed acceptance and evaluation criteria for each component.

The I&E Guidelines in MRP-227 build on the existing ASME Code, Section XI inspections to create comprehensive inspection recommendations for aging degradation in reactor internals [7,8]. The fundamental goal of any inspection under the ASME Code, Section XI program [9] is to identify relevant conditions that require further action. Article IWA-9000 of the ASME Code defines relevant condition as follows:

> Relevant Condition – A condition observed during a visual examination that requires supplemental examination, corrective measure, correction by repair/replacement activities, or analytical evaluation.[2]

The ASME Code [9] defines acceptance standards in IWB-3400, which are used to determine whether an observed condition is acceptable for service or is a relevant condition. The inspection standards used to define relevant condition are based on generic analysis that provides a high level of assurance of satisfactory function. Acceptance standards for ASME Examination Categories B-N-2 (Welded Core Support Structures and Interior Attachments to Reactor Vessels) and B-N-3 (Removable Core Support Structures) are provided in IWB-3520. While these acceptance standards are appropriate for the ASME Code Section XI inspections, including those that are specifically highlighted in MRP-227 as existing reactor internals inspection requirements, they are not fully applicable to the MRP-227 new inspection recommendations. In particular, the referenced linear flaw standards of IWB-3510 are intended to guard against propagation of cracks through the reactor pressure vessel are not meaningful when applied to the removable internals components.

2. ASME (Article IWA-9000) uses this definition for visual examinations. The definition has been expanded here to include all inspections conducted under the I&E Guidelines.

MRP-227 provides lists of components with specific inspection recommendations. The components are divided into three basic categories based on the recommendation:

- Primary Components – Inspection of the Primary Components is required in the I&E Guidelines. In general, these inspections must be conducted early in the license extension period. Every plant needs to have acceptance criteria for Primary Components included in their aging management program.

- Expansion Components – The MRP -227 defines criteria for the results of Primary Component inspections that will trigger inspection of the Expansion Components. Although the MRP-227 requirements allow a time delay for mobilization of equipment and resources, acceptance criteria for the expansion components will also be required.

- Existing Components – There are established acceptance criteria for the ASME Code, Section XI and other exams in this category. The PWROG may want to review the current field practice on these inspections as part of the implementation evaluations to ensure a consistent approach.

Each of these components also requires some acceptance standard for evaluating the relevant condition. For many general condition monitoring visual examinations (VT-3), MRP-227 identifies specific conditions that supplement the list of relevant conditions provided in IWB-3520.2. However, there is no comprehensive listing of acceptance standards in MRP-227. The general standards for determining relevant conditions in reactor internals inspections are provided in MRP-228 [10]:

- Welds – Cracks, or indications that exhibit characteristics of cracking, are considered relevant.

- Components – Cracking or other significant degradation that could impair the ability of the component to perform its design function is considered relevant.

Once a relevant condition has been identified, evaluation is required to assess the ability of the degraded component to continue to perform the design function without interfering with the function of the system. The evaluation of suitability for continued service will be, in general, an extension of the analysis used to define relevant condition. Although specific acceptance criteria and evaluation procedures need not be fully developed at the time that the I&E Guidelines are submitted to the NRC, it is critical that there be a clear path to successful implementation. The scope of this PWROG Reactor Internals Core Team (RICT) Project Authorization is to provide this path by defining the process for developing these acceptance criteria and evaluation procedures. The process will be defined in the following terms:

- Evaluation Methodology – The procedures and criteria to be used by the engineering staff to evaluate relevant conditions. This includes:
  - Demonstration of functionality of the current configuration
  - Establishment of a re-inspection frequency of one or more refueling cycles
  - An engineering basis for repair/replacement/mitigation options

- Acceptance Criteria – The criteria against which the need for corrective action will be evaluated. The acceptance criteria should ensure that the intended functions of the particular structure and component are maintained under all current licensing basis design conditions during the period of extended operation.[3]

In some cases, it will be feasible to avoid plant-specific evaluations by adopting generic standards for acceptance that will ensure compliance with the accepted industry requirements.

- Generic Acceptance – Disposition of a relevant condition based on a generic implementation of the evaluation methodology for an NSSS design or other plant grouping.

The original MRP request forwarded to the PWROG RI-CT was for generic acceptance criteria. However, as a prerequisite to developing the generic criteria, the evaluation methodology must be defined. The development of evaluation methodologies also poses significant implementation issues that need to be considered by the PWROG. Therefore, the Phase I program summarized in this document outlines the evaluation methodology for components to be inspected under the proposed I&E Guidelines and identifies analyses that may be conducted on a generic basis. The Phase I program defines potential scope for additional PWROG projects to support the development of evaluation methodologies and generic acceptance criteria.

3. Definition based on [12] Aging Management Program (AMP), Element 6.

3 ANALYSIS PROCESS

The goal of this effort is to define the process to be used in the degradation evaluation procedures and the necessary data requirements to perform the evaluation of each Primary and Expansion component such that engineering organizations follow a consistent approach that is documented and approved. The proposed guidelines and methodologies will be listed in outline format and illustrated in a logic chart showing potential evaluation and disposition options for the relevant condition.

Both AREVA and Westinghouse assembled teams of experts to develop recommendations. Although the procedural details and reporting format of these efforts are unique to the vendors, the basic process followed the same general steps:

1. Review MRP-227 degradation modes and inspection recommendations.
2. Determine component function.
3. List potential inspection outcomes and observable effects.
4. Identify potential failure mechanisms and effects.
5. Outline methodology to evaluate potential inspection observations.
6. Determine data requirements for inspection.
7. Consider potential for vendor-specific generic analysis.

The form and structure of the evaluation methodology is determined by the type of inspection recommended.

3.1 PHYSICAL MEASUREMENTS

Physical measurements are employed to characterize changes in component dimension. For example, measurements of internals hold-down spring height are used to evaluate loss of core hold-down forces due to wear or stress relaxation. In this case, the required hold-down forces are a design requirement and generic or plant-specific acceptance criteria should be established prior to the inspection.

No specific action to establish general acceptance criteria was required under this task. The required methodologies will simply note the existence of relevant design requirements for the affected components.

3.2 GENERAL CONDITION

There is a heavy reliance on general condition monitoring within the I&E recommendations. Although these recommendations refer to the VT-3 level visual inspections, the guidelines generally provide a description of the expected degradation. Under the VT-3 procedure, any observation of degradation is considered to be a relevant condition and is reported to engineering for evaluation. Generally, engineering will review the observation to either confirm the relevant condition or disposition it as a visual anomaly.

Due to the qualitative nature of the VT-3 examination, it is difficult to define quantitative evaluation and acceptance criteria. In some cases, VT-3 examinations have been specified for redundant components, where multiple failures are required to impair the functionality of the system. In other cases, VT-3 has been specified where the first signs of degradation are expected to be visual (e.g., wear). In any case, it is prudent engineering practice to anticipate the range of possible visual observations and define resolution strategies.

A process identifying potential relevant conditions arising from each recommended VT-3 examination and defining resolution strategies is required for each VT-3 examination. This process may take the form of a failure modes and effects analysis (FMEA).

3.3 ULTRASONIC TESTING (BOLTS)

Within the I&E Guidelines, ultrasonic testing (UT) examinations are conducted to identify failed bolts. All bolts with positive indications of cracking are assumed to be failed. The bolting systems in the internals are generally highly redundant. Acceptance criteria are based on minimum bolting patterns that guarantee structural stability through both normal operation and design basis transients. To establish an appropriate inspection interval, the current distribution of unfailed bolts must contain sufficient margin to demonstrate that the number of anticipated failures will not cause the distribution to fall below the minimum pattern. This will require either a historical analysis of bolt failure rates or a detailed model of bolt failure mechanisms.

The use of minimum bolting patterns as acceptance criteria that allow individual bolt failures is established in the industry. The PWROG (from prior Westinghouse Owners Group [WOG] efforts) has developed minimum bolting patterns for baffle-to-former bolting for Westinghouse reactor internals designs. The PWROG has supported development of similar strategies for core barrel bolt inspections in Babcock & Wilcox (B&W) plants. The methodology for performing a minimum bolting pattern or similar strategy is beyond the scope of the current task.

3.4 VISUAL CRACKING

Visual examinations to identify cracking are generally recommended where intergranular stress corrosion cracking (IASCC), stress corrosion cracking (SCC), or fatigue is identified as the cracking mechanism. The initial visual examinations in the B&W plants are all based on VT-3 requirements. The VT-3 exams were deemed to be adequate because the structures are relatively flaw tolerant. Appropriate follow-on actions might include EVT-1, UT, or eddy-current testing (ET) exams to determine flaw size. These options would be identified as part of the recommended procedure for resolution of the original VT-3 observation. Consistent with the current state-of-the-art within the Boiling Water Reactor Vessel and Internals Project (BWRVIP), EVT-1 has been identified as the appropriate visual examination procedure for several of the Combustion Engineering (CE) and Westinghouse components. The EVT-1 examination can produce flaw size information that would lead directly to a fracture mechanics evaluation. UT or ET may also be used as enhanced or supplemental examinations, where appropriate.

General fracture mechanics procedures for calculating critical flaw sizes and growth rates are described within the I&E Guidelines. In order to apply these procedures, the appropriate irradiation history, loading conditions and stress intensity solutions must be identified. These factors are all dependent on both the flaw location and the plant design.

This task outlines specific fracture mechanics analysis requirements for each of the visual EVT-1 examinations included in the Primary and Expansion tables of the I&E Guidelines.

3.5 VISUAL OTHER

A VT-1 level examination was identified to examine potential swelling-related distortion in some welded core shroud structures originally designed by CE. The intention of the examination is to provide semi-quantitative data that can be used to evaluate the overall level of swelling in the structure. The original functionality analysis, which is known to be conservative, predicted large gap openings at specific locations in the shroud structure.

These examinations are meant to provide an early warning of swelling in the structure. Evaluation of this swelling-related data will require additional sensitivity studies to relate the swelling level to the predicted distortion and gap opening in the structure. Sensitivity studies are currently being considered by the MRP and are beyond the scope of this project.

3.6 FATIGUE (QUALIFY BY TIME-LIMITED AGING ANALYSIS [TLAA])

Four component items in the CE design are included in the list of Primary components solely due to concerns about fatigue. Due to the plant-specific nature of the TLAA required for license extension programs, fatigue analysis was not included in the MRP functionality analysis. It is considered a high probability that TLAA will demonstrate a negligible probability of fatigue crack initiation in these components. However, pending resolution by TLAA, these component items are included in the Primary Component list. Acceptance criteria for these four fatigue-related items are not included in this task.

3.7 RESOLUTION BY ANALYSIS

Three Expansion component items in the B&W designs have been designated for resolution by analysis. Inspection of these three components is considered to be impractical due to issues of accessibility. Therefore, should concerns about the integrity of these components be triggered by observations in the associated Primary components, no inspection is required. Resolution would require either detailed analysis or replacement.

Acceptance criteria for these three Expansion components are not included in this task. A separate project authorization to support additional analysis of these three B&W components may be proposed to the PWROG.

4 RECOMMENDATION FORMAT

Recommendations for each Primary and Expansion item were prepared by Westinghouse and AREVA after consulting with appropriate experts and expert panels. The basic requirements for this study were mutually agreed upon between the vendors and defined in the original project authorization. The internal processes employed to complete these studies were specific to the vendor. The results of the Westinghouse and AREVA analyses are provided in Appendix A (B&W plants), Appendix C (CE plants), and E (Westinghouse plants). Although there are superficial differences in the format in which the results are presented by each vendor, the underlying structure of the information is the same.

4.1 IDENTIFICATION AND EXAMINATION RECOMMENDATIONS

Component identification and inspection recommendations for both the Primary and Expansion component items were originally provided in MRP-227. Relevant rows from MRP-227 Tables 4.1 and 4.4 were reproduced as part of the AREVA analysis of component items from the B&W plants as shown in Appendix A. Corresponding information on CE plants was extracted from MRP-227 Tables 4.2 and 4.5 and integrated into the Westinghouse data forms shown in Appendix C. A similar process was used to extract information on Westinghouse plants from MRP-227 Tables 4.3 and 4.6 for the data for the Westinghouse data forms in Appendix E.

4.2 COMPONENT FUNCTION

To facilitate consideration of consequence of failure, both vendor processes required a brief summary of the component function. More extensive function descriptions for the reactor internals components were originally compiled in support of the Consequence Analysis performed for the Issue Management Tables. These summaries are included in MRP-156 [11]. The component function summaries provided in Appendices A, C, and E are meant only to provide perspective for the analysis recommendations.

4.3 INSPECTION OUTCOMES

The inspection recommendations provided in MRP-227 are based on the identification of specific aging degradation mechanisms. The expert teams were asked to consider the information potentially generated by the inspection. The AREVA evaluations are summarized in Appendix A under the headings "Observable Effects" and "Possible Outcomes." The Westinghouse evaluations are summarized in Appendices C and E under the headings "Observable Effect," "Failure Mechanism," "Failure Effect," and "Failure Criteria."

4.4 METHODOLOGY AND DATA REQUIREMENTS

The objective of this study was to define methodologies and data requirements for analysis and acceptance of degraded components identified in the MRP-227 recommended inspection. The AREVA format provides this information under a single heading. The Westinghouse format divides the methodology into four separate subheadings: "Goal," "Data Requirements," "Analysis," and "Acceptance Criteria." For the Westinghouse sections, unless otherwise indicated by specific terminology, "operating loads" refers to any loads generated under normal operating conditions.

4.5 RECOMMENDED APPROACH

In general, implementation of the methodology and data requirement recommendations can require a combination of plant-specific and cooperative actions. Because the MRP-227 recommendations tend to be very design specific, generic actions that apply to the entire PWR fleet are not expected. However, it may be possible to define cooperative activities relevant to rational subgroups such as plants designed by a single vendor.

5 B&W PLANT DESIGN RESULTS

5.1 METHODOLOGIES

A summary of the guidelines and methodology for determining acceptance criteria and the needed data requirements suggested by the expert panel for each of the Primary and Expansion component items identified in MRP-227 for the operating B&W-design reactor vessel internals is provided in Appendix A.

Appendix B provides logic charts illustrating the evaluation path and potential disposition options for relevant inspection conditions for each of the Primary and Expansion component items identified in MRP-227.

5.2 GENERIC ACCEPTANCE CRITERIA

Each of the Primary and Expansion component items was evaluated to determine those for which analyses would be practicable to develop generic acceptance criteria. Differing unit loads, transients, materials, etc. were considered in identifying those component items that could be analyzed on a generic basis. The AREVA analysis indicates actions that might be used to define nondestructive examination (NDE) acceptance standards and actions that could support analytical evaluations. Table 5-1 below provides the results of this effort.

Table 5-1 Applicability of Potential Generic Acceptance Criteria for B&W-Design Primary and Expansion Component Items

Component Item	Develop Generic Acceptance Criteria?(1)		Comments
	Analytical	NDE Standard	
Primary Items			
Plenum Cover Assembly & Core Support Shield Assembly Plenum cover weldment rib pads Plenum cover support flange CSS top flange	Yes	Yes	
Core Support Shield Assembly CSS cast outlet nozzles (Applicable to ONS-3 and DB only)(2)	Yes	Yes	Analytical efforts applicable to both ONS-3 and DB could be performed, although unit-specific analytical efforts may provide additional margin for one or both units.
Core Support Shield Assembly CSS vent valve discs(2)	Yes	Yes	A unit-specific bypass analytical effort is required for DB, since the number of vent valves is different.
Core Support Shield Assembly CSS vent valve top retaining ring CSS vent valve bottom retaining ring CSS vent valve disc shaft or hinge pin	Yes	Yes	Analytical efforts could be performed on a generic basis for all B&W units, although unit-specific analytical efforts may provide additional margin (particularly for DB).
Core Support Shield Assembly Upper core barrel (UCB) bolts and their locking devices	No	Yes	Due to the variations in bolt materials used and loadings among the units, unit-specific analytical efforts are required. The generic efforts have already been completed in the PWROG PA-MSC-350 work.
Core Barrel Assembly Lower core barrel (LCB) bolts and their locking devices	No	Yes	Due to the variations in bolt materials used and loadings among the units, unit-specific analytical efforts are required. The generic efforts have already been completed in the PWROG PA-MSC-350 work.
Core Barrel Assembly Baffle-to-former bolts	Yes	N/A	There are two designs (ONS-1 and TMI-1 are one design, and the other five units are the second design).
Core Barrel Assembly Baffle plates	Yes	Yes	

Table 5-1 (cont.) Applicability of Potential Generic Acceptance Criteria for B&W-Design Primary and Expansion Component Items

Component Item	Develop Generic Acceptance Criteria?(1) Analytical	NDE Standard	Comments
Core Barrel Assembly Locking devices, including locking welds, of baffle-to-former bolts and internal baffle-to-baffle bolts	N/A	Yes	There are two designs (ONS-1 and TMI-1 are one design and the other five units are the second design).
Lower Grid Assembly Alloy X-750 dowel-to-guide block welds	Yes	Yes	
Incore Monitoring Instrumentation (IMI) Guide Tube Assembly IMI guide tube spiders IMI guide tube spider-to-lower grid rib section welds	Yes	Yes	
		Expansion Items	
Upper or Lower Grid Assembly Alloy X-750 dowel-to-upper grid fuel assembly support pad welds or Alloy X-750 dowel-to-lower grid fuel assembly support pad welds	Yes	Yes	
Control Rod Guide Tube Assembly CRGT spacer castings	No	Yes	Reactivity analyses are dependent upon fuel loading and must be performed on a unit-specific basis.
Core Barrel Assembly Upper thermal shield (UTS) bolts and their locking devices	Yes	Yes	Analytical efforts for the UTS bolt failures could be performed on a generic basis for all units except TMI-1, although use of unit-specific loadings could reduce the conservatism for some units.
Core Barrel Assembly Surveillance specimen holder tube (SSHT) studs/nuts (CR-3) or bolts (DB) and their locking devices	No	No	There are only two units: one has studs/nuts and the other has bolts.
Core Barrel Assembly Core barrel cylinder (including vertical and circumferential seam welds) Former plates	Yes	N/A	

Table 5-1 (cont.) Applicability of Potential Generic Acceptance Criteria for B&W-Design Primary and Expansion Component Items

Component Item	Develop Generic Acceptance Criteria?(1) Analytical	NDE Standard	Comments
Core Barrel Assembly Baffle-to-baffle bolts Core barrel-to-former bolts	Yes	N/A	There are two designs (ONS-1 and TMI-1 are one design, and the other five units are the second design).
Core Barrel Assembly Locking devices, including locking welds, for the external baffle-to-baffle bolts and core barrel-to-former bolts	Yes	N/A	
Lower Grid Assembly Lower grid fuel assembly support pad items: pad, pad-to-rib section welds, Alloy X-750 dowel, cap screw, and their locking welds (Note: The pads, dowels, and cap screws are included because of TE/IE of the welds.)	Yes	Yes	
Lower Grid Assembly Lower grid shock pad bolts and their locking devices (TMI only)	No	No	TMI unit-specific
Lower Grid Assembly Lower thermal shield studs/nuts or bolts (LTS) and their locking devices	No	Yes	Due to the variations in stud/nut or bolt materials used and loadings among the units, unit-specific analyses are required.
Flow Distributor Assembly Flow distributor (FD) bolts and their locking devices	Yes	Yes	Analytical efforts for the FD bolts could be performed on a generic basis (for all units except TMI-1), although unit-specific analyses could decrease the conservatism for some units.

Notes:

1. Analytical efforts include finite element analysis or fracture mechanics analysis. An NDE inspection standard contains examples of acceptable and unacceptable visual indications or ultrasonic testing flaw sizes.
2. These items may potentially be removed from examination if a records search identifies that actual material heats used for fabrication could be screened out as not being susceptible to the thermal aging degradation mechanism.

6 COMBUSTION ENGINEERING AND WESTINGHOUSE PLANT DESIGN RESULTS

6.1 METHODOLOGIES

Datasheets outlining the guidelines and methodology for determining acceptance criteria and the needed data requirements suggested by the expert panel for each of the Primary and Expansion component items in CE and Westinghouse plants are provided Appendices C and E. CE recommendations are contained in Appendix C. Westinghouse plant recommendations are contained in Appendix E.

Appendices D and F provide logic charts illustrating the evaluation path and potential disposition options for relevant inspection conditions for each of the Primary and Expansion component items identified in MRP-227.

6.2 GENERIC ACCEPTANCE CRITERIA

Comments on the analysis approach for each component are included in the final section of the datasheets included in Appendices C and E. These recommendations, which include any actions that might be taken on a generic basis, are summarized in Tables 6-1 and 6-2.

Table 6-1 Applicability of Potential Generic Acceptance Criteria for CE-Design Primary and Expansion Component Items

CE Component	Approach
CE-ID: 1 Core Shroud Bolts	No generic effort required. Only two plants are affected.
CE-ID 1.1 Barrel Shroud Bolts	No generic effort required. Only two plants are affected.
CE-ID: 1.2 Core Support Column Bolts	Generic program to share first-of-a-kind effort. (See W-ID: 2.1) • Pilot analysis of lower support structure to identify critical issues. • Expect final acceptance based on plant-specific analysis.
CE-ID: 2 Core Shroud Plate-Former Plate Weld	Expect calculation to be plant specific. • Define general load conditions at weld seams. • Define K-solution for loading at weld seams.
CE-ID: 2.1 Remaining Axial Welds	Plant-specific analysis. • Require flaw tolerance handbook/methodology based on flaw location and direction.
CE-ID: 3 Shroud Plates (Full Height)	No generic analysis: Only one utility with this design.
CE-ID: 3.1 Remaining Axial Welds; Ribs and Rings	No generic analysis: Only one utility with this design.
CE-ID: 4 Core Shroud Assembly (Bolted)	FMEA should address plant-specific practices and priorities. Some generic work possible to outline issues and options to be addressed in FMEA.
CE-ID: 5 Core Shroud Assembly (Welded)	Generic efforts to support inspection. • Extension of MRP model to look at relationship between swelling and deformation at seam. • Guideline for issues to be addressed in plant-specific FMEA.
CE-ID: 6 Upper Core Support Barrel Flange Weld	Plant-specific analysis. • Similar to Ginna pilot plant experience. (See W-ID: 3)
CE-ID: 6.1 Lower Core Barrel Flange	Plant-specific analysis. • Require flaw tolerance handbook/methodology based on flaw location and direction. • MRP-210 may have limited relevance.

Table 6-1 (cont.)	**Applicability of Potential Generic Acceptance Criteria for CE-Design Primary and Expansion Component Items**
CE Component	**Approach**
CE-ID: 6.2 Remaining Core Barrel Assembly Welds	Plant-specific analysis. (See item CE-ID: 6.1) • Require flaw tolerance handbook/methodology based on flaw location and direction. • MRP-210 may have limited relevance.
CE-ID: 7 Core Support Barrel Lower Flange Weld	TLAA (plant specific) Potential flaw analysis if inspection required. • Require flaw tolerance handbook/methodology based on flaw location and direction. • MRP-210 may have limited relevance.
CE-ID: 8 Core Support Plate	TLAA (plant specific)
CE-ID: 9 Upper Fuel Alignment Plate	TLAA (plant specific – applies to one utility)
CE-ID: 10 Instrument Guide Tubes	Pass/Fail inspection with established minimum number of instrumentation tubes. Based directly on plant specifications.
CE-ID: 10.1 Remaining Instrument Guide Tubes	Pass/Fail inspection with established minimum number of instrumentation tubes. Based directly on plant specifications. (See CE-ID: 10)
CE-ID: 11 Deep Beams	TLAA (plant specific – applies to one utility)

Table 6-2 Applicability of Potential Generic Acceptance Criteria for Westinghouse-Design Primary and Expansion Component Items	
Westinghouse Component	**Approach**
W-ID: 1 Control Rod Guide Tube (CRGT) Assembly Guide Plates (Cards)	Generic work ongoing under PWROG program • Validate and/or modify linear volumetric wear rate model. Potential extension • Alternative justification that allows wear through ligament in one or more cards.
W-ID: 2 CRGT Lower Flange Weld	Plant-specific analysis due to large variety of sizes and designs. There may be some potential for smaller plant groupings.
W-ID: 2.1 Lower Support Column Bodies (Cast)	Generic program to share first-of-a-kind effort. • Pilot analysis of lower support structure to identify critical issues. • Expect final acceptance based on plant-specific analysis.
W-ID: 2.2 Bottom-mounted Instrumentation Column Bodies	Pass/Fail inspection with established minimum number of instrumentation tubes. Based directly on plant specifications.
W-ID: 3 Upper Core Barrel Flange Weld	Plant-specific analysis. • Ginna provides pilot plant experience in the creation of generic acceptance criteria. • May be able to group plants by design.
W-ID: 3.1 Other Core Barrel Welds	Plant-specific analysis. • Require flaw tolerance handbook/methodology based on flaw location and direction. • MRP-210 may have limited relevance.
W-ID: 3.2 Lower Support Column Bodies (Non-cast)	Generic program to share first-of-a-kind effort. (See W-ID: 2.1) • Pilot analysis of lower support structure to identify critical issues. • Expect final acceptance based on plant-specific analysis.
W-ID: 4 Baffle-edge Bolts	FMEA should address plant-specific practices and priorities. Some generic work possible to outline issues and options to be addressed in FMEA.
W-ID: 5 Baffle-former Bolts	Generic work completed in previous PWROG program.
W-ID: 5.1 Barrel-former Bolts	Generic work completed in previous PWROG program.

Table 6-2 (cont.) Applicability of Potential Generic Acceptance Criteria for Westinghouse-Design Primary and Expansion Component Items

Westinghouse Component	Approach
W-ID: 5.2 Lower Support Column Bolts	Generic program to share first-of-a-kind effort. (See W-ID: 2-1) • Pilot analysis of lower support structure to identify critical issues. • Expect final acceptance based on plant-specific analysis.
W-ID: 6 Baffle-former Assembly	FMEA should address plant-specific practices and priorities. Some generic work possible to outline issues and options to be addressed in FMEA.
W-ID: 7 Internals Hold-down Spring	Value determined by plant-specific design requirements.
W-ID: 8 Thermal Shield Flexures	None: Plant-specific analysis.

7 REFERENCES

1. Materials Reliability Program: Pressurized Water Reactor Internals Inspection and Evaluation Guidelines (MRP-227-Rev. 0). EPRI, Palo Alto, CA: 2008. 1016596.

2. Materials Reliability Program: Screening, Categorization, and Ranking of B&W-Designed PWR Internals Component Items (MRP-189 Rev. 1). EPRI, Palo Alto, CA: 2009. 1018292.

3. Materials Reliability Program: Failure Modes, Effects, and Criticality Analysis of B&W-Designed PWR Internals (MRP-190). EPRI, Palo Alto, CA: 2006. 1013233.

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

5. Materials Reliability Program: Functionality Analysis for B&W Representative PWR Internals (MRP-229 Rev. 1). EPRI, Palo Alto, CA: 2009. 1019090.

6. Materials Reliability Program: Functionality Analysis for Westinghouse and Combustion Engineering Representative PWR Internals (MRP-230). EPRI, Palo Alto, CA: 2008. 1016597.

7. Materials Reliability Program: Aging Management Strategies for B&W PWR Internals (MRP-231 Rev. 1). EPRI, Palo Alto, CA: 2009. 1019092.

8. Materials Reliability Program: Aging Management Strategies for Westinghouse and Combustion Engineering PWR Internals (MRP-232). EPRI, Palo Alto, CA: 2008. 1016593.

9. ASME Boiler and Pressure Vessel Code, Section XI, Division 1, "Rules for Inservice Inspection of Nuclear Power Plant Components," American Society of Mechanical Engineers, New York, NY: 2001 Edition, Plus 2003 Addenda, or later.

10. Materials Reliability Program: Inspection Standard for PWR Internals (MRP-228). EPRI, Palo Alto, CA: 2009. 1016609.

11. Materials Reliability Program: Pressurized Water Reactor Issue Management Table, PWR-IMT Consequence of Failure (MRP-156). EPRI, Palo Alto, CA: 2005. 1012110.

12. U.S. Nuclear Regulatory Commission Report, NUREG-1801, Vol. 1, Rev. 1, "Generic Aging Lessons Learned (GALL) Report," September 2005.

13. Westinghouse Report, WCAP-15030-NP-A, Rev. 0, "Westinghouse Methodology for Evaluating the Acceptability of Baffle-Former-Barrel Bolting Distributions under Faulted Load Conditions," March 2, 1999.

APPENDIX A
B&W DESIGN PRIMARY AND EXPANSION COMPONENT ITEM ACCEPTANCE CRITERIA METHODOLOGY AND DATA REQUIREMENTS

A.1 PRIMARY COMPONENT ITEMS

Acceptance criteria methodology and data requirements for each of the Primary component items are summarized in this appendix. A separate sub-section is provided for each component item using the following format:

- Primary component item information extracted directly from Table 4-1 of MRP-227.
- This information is in tabular form and contains the item name, unit applicability, failure effect, failure mechanism(s), expansion link(s), examination method, examination frequency, and examination coverage.
- Component item function(s), including whether or not it has a core support safety function.
- Observable effect(s).
- Methodology for development of acceptance criteria.
- Data requirements for development of acceptance criteria.
- Existing documents (e.g., PWROG or AREVA).

Note that repairs or replacements are potential options for various components that are not summarized in this appendix, but are included in Appendix B.

Core Clamping Items

Item	Applicability	Effect (Mechanism)	Expansion Link	Examination Method/Frequency	Examination Coverage
Plenum Cover Assembly & Core Support Shield Assembly Plenum cover weldment rib pads Plenum cover support flange CSS top flange	All plants	Loss of material and associated loss of core clamping pre-load (Wear)	None.	One-time physical measurement no later than two refueling outages from the beginning of the license renewal period. Perform subsequent visual (VT-3) examination on the 10-year ISI interval.	Determination of differential height of top of plenum rib pads to reactor vessel seating surface, with plenum in reactor vessel. See MRP-231 Figure 3-1.

The potential exists for this to be a high wear location with subsequent reduction or loss of core clamping pre-load. This is a unique configuration with the B&W units and no known OE history indicating wear currently exists.

<u>Component Item Function</u>

The purpose of the core clamping load is to stabilize and significantly restrict the rigid body pendulum motion of the core support and plenum assemblies. In other words, the clamping action prevents rigid body rotation at the interface area. The clamping action does not have a direct core support safety function. Loss of clamping would undoubtedly lead to core barrel motions that would eventually lead to a reactor shut down.

<u>Observable Effects:</u>

A one-time physical measurement is to be obtained. The interference measurement to the nearest 0.001 inch is to be recorded at eight locations at approximately 45-degree intervals. Subsequent follow-up visual (VT-3) examinations are to be obtained during the ASME Code B-N-3 10-year ISI activities.

ONS has performed physical measurements on a unit-specific basis and no measurable wear has been observed. TMI-1 plans to obtain the data during the Fall 2009 outage. The remaining units have not completed the measurement to date.

The physical measurement is performed to determine the differential height from the top of the plenum cover assembly weldment rib pads to the reactor vessel seating surface. The measurement is the stack-up of the core support flange and the plenum support area versus the reactor vessel support ledge. The measurement must be taken without fuel in the reactor to eliminate the effect of the fuel hold-down springs. This interference fit was measured during original site assembly and also as-built measurements were taken of the piece parts after fabrication. The interference fit ranged between 0.008 and zero for the operating units.

Identification of an unacceptable condition is the precursor to perform follow-on investigations or analysis to establish the extent of degradation. The subsequent VT-3 inspections are to look for wear on the top and bottom surface of the CSS top flange, bottom surface of the plenum cover assembly support flange, and the top surface of the plenum cover assembly weldment rib pads.

Possible Examination Outcomes:

- No wear observed (data falls within scatter of original measurements)
- Wear of some extent is observed (one location, several locations, etc.)
- Other relevant conditions identified (e.g., missing rib pad)

Methodology and Data Requirements:

The acceptance criterion is based on engineering judgment, and is defined as a reduction of no greater than 0.004 inch compared to the original as-built data. The criterion of 0.004 inch reduction in interference is not be construed as an indication of inadequate clamping, but an indication that the surface conditions have changed since the unit was put into service. Additional inspection, using VT-3 for example, will be required to verify that wear is actually occurring.

The general analytical methodology to be used for determining the wear acceptance criterion involves the following steps and inputs:

- Determine the minimum core clamping preload required
  - This requires the differential pressure distribution on the core support cylinder due to reactor coolant flow
  - This also includes evaluation for different minimum pre-loads versus time at different operating conditions
- Determine an uncertainty margin, which includes both input pressure differential and measurement error
- Determine how the clamp load varies with operating conditions
- Develop a wear estimate for time from discovery of wear to time for required remediation

The general methodology to be used for VT-3 acceptance criteria for these component items will be development of an NDE inspection standard that contains examples of acceptable and unacceptable visual indications and mockups for the VT-3 inspection of wear. Input information needed includes:

- Identification of the most likely signs and locations of wear that can be inspected
- Identification of what visual examination wear indications are considered rejectable and would require additional dimensional examination and evaluation

- Identification of any additional examination results that are anticipated
  - Identify general acceptance criteria for the additional items expected to be in the VT-3 examination field of vision

Analytical efforts could be performed on a generic basis. The NDE inspection standard could also be developed generically.

Existing Documentation:

- In the design stage, a design clamping load at operational conditions was established based on the flow uplift loading and horizontal forces developed by various pump combinations. An additional study in 1974 indicated that the clamping value was marginal using the reactor vessel stud pre-load identified in the reactor vessel instruction manual. A new pre-load was incorporated into the design, which provided satisfaction of the criteria at operating conditions but indicated there may be some loss of clamping at low temperatures (approximately < 300°F). It was verified by observation that the potential loss of pre-load was not leading to short-term degradation of the clamping surfaces.
- No acceptable evaluation or analysis has been completed to date for determining a re-inspection schedule.
  - A wear rate estimate would be needed to project the wear over one cycle or more for continued operation.

What observations trigger examination into the Expansion category?

There are no expansion component item links for this examination.

Core Support Shield Cast Outlet Nozzles

Item	Applicability	Effect (Mechanism)	Expansion Link	Examination Method/Frequency	Examination Coverage
Core Support Shield Assembly CSS cast outlet nozzles	ONS-3, DB	Cracking (TE), including the detection of surface irregularities, such as damaged or fractured material	CRGT spacer castings	Visual (VT-3) examination during the next 10-year ISI. Subsequent examinations on the 10-year ISI interval.	100% of accessible surfaces. See MRP-231 Figure 3-9.

CSS cast outlet nozzles are subject to thermal aging embrittlement, which if a flaw would be present and they are subjected to loading that exceeds the materials degraded fracture toughness, such a condition could potentially lead to cracking. There is no known history of OE identifying cracking of CASS material in PWR reactor vessel internals applications.

These items may potentially be removed from examination if a records search identifies that actual material heats used for fabrication could be screened out as not being susceptible to the thermal aging degradation mechanism.

Component Item Function

Degradation of the outlet nozzles could result in a core cooling issue under normal operation because of increased core bypass flow and a reduction in margin to DNB (see MRP-157). The outlet nozzles do not have a core support safety function; however, they do have a safety function to control bypass around the core during a loss-of-coolant-accident (LOCA).

Observable Effects:

A visual (VT-3) examination of the outlet nozzles is to be performed. Subsequent visual examinations are to be performed during the ASME Code B-N-3 10-year ISI activities.

The outlet nozzles are being examined to detect damage either in the form of a precursor to the loss of material or a piece or section of the material that has fractured and is currently missing. The location that potentially contains the highest tensile stresses is near the heat-affected-zone (HAZ) of the outlet nozzle weld-to-core support shield cylinder.

Possible Examination Outcomes:

- No relevant conditions identified
- One or more areas are identified with crack-like indications
- One or more areas are identified with missing material

Methodology and Data Requirements:

The general analytical methodology to be used for acceptance criteria for the CSS cast outlet nozzles involves the following steps and inputs:

- Perform a bypass analysis to justify that sufficient DNB exists in the degraded condition
- A VT-1, ET, or UT examination may be needed to determine if flaws are emanating from the location where missing material may be identified

The general methodology to be used for acceptance criteria for these component items will be development of an NDE inspection standard that contains examples of acceptable and unacceptable visual indications and mockups for the VT-3 inspection of cracking. Input information needed includes:

- Identification of the most likely locations of surface irregularities, such as damaged or fractured material
- Identification of what visual examination indications are considered rejectable and would require additional examination and evaluation

Analytical efforts applicable to both ONS-3 and DB could be performed, although unit-specific analytical efforts may provide additional margin for one or both units. The NDE inspection standard could also be developed generically.

Existing Documentation:

- CLB loadings (normal and faulted condition) are available, but a records search may need to be performed to identify them
- No acceptable evaluation or analysis has been completed to date for determining a re-inspection schedule

What observations trigger examination into the Expansion category?

- Confirmed evidence of relevant conditions for a single CSS cast outlet nozzle shall require expansion to the CRGT spacer castings by the completion of the next refueling outage

Core Support Shield Vent Valve Discs

Item	Applicability	Effect (Mechanism)	Expansion Link	Examination Method/Frequency	Examination Coverage
Core Support Shield Assembly CSS vent valve discs (Note 1)	All plants	Cracking (TE), including the detection of surface irregularities, such as damaged or fractured material	CRGT spacer castings	Visual (VT-3) examination during the next 10-year ISI. Subsequent examinations on the 10-year ISI interval.	100% of accessible surfaces. (See BAW-2248A, page 4-3 and Table 4-1.) See MRP-231 Figures 3-10 and 3-11.

Notes:
1. A verification of the operation of each vent valve shall also be performed through manual actuation of the valve. Verify that the valves are not stuck in the open position and that no abnormal degradation has occurred. Examine the valves for evidence of scratches, pitting, embedded particles, variation in coloration of the seating surfaces, cracking of lock welds and locking cups, jack screws for proper position, and wear. The frequency is defined in each unit's technical specifications or in their pump and valve in-service test programs (see AREVA doc. BAW-2248A, page 4-3, and Table 4-1).

Vent valve discs are subject to thermal aging embrittlement, which if a flaw would be present and they are subjected to loading that exceeds the materials degraded fracture toughness, such a condition could potentially lead to cracking. There is no known history of OE identifying cracking of CASS material in PWR reactor vessel internals applications.

These items may potentially be removed from examination if a records search identifies that actual material heats used for fabrication could be screened out as not being susceptible to the thermal aging degradation mechanism.

Component Item Function

Vent valves are passive devices that have no function during normal operation. The vent valve discs do not have a core support safety function; however, they do have a safety function in that degradation of the vent valve discs, which would prevent the vent valve from opening, could result in loss of the vent valve function during a large break loss-of-coolant-accident (LOCA).

Observable Effects:

A visual (VT-3) examination of the vent valve discs is to be performed. Subsequent visual examinations are to be performed during the ASME Code B-N-3 10-year ISI activities.

The vent valve discs are being examined to detect damage either in the form of a precursor to the loss of material, or, a piece or section of the material that has fractured or is currently missing.

Possible Examination Outcomes:

- No relevant conditions identified

- One or more areas are identified with crack-like indications
- One or more areas are identified with missing material

Methodology and Data Requirements:

The general analytical methodology to be used for acceptance criteria for the CSS vent valve discs involves the following steps and inputs:

- Perform a bypass analysis to justify that sufficient DNB exists in the degraded condition

- Perform an analysis to show that failure of the vent valve disc will not result in loss of function

- A VT-1, ET, or UT examination may be needed to determine if flaws are emanating from the location where missing material may be identified

The general methodology to be used for acceptance criteria for these component items will be development of an NDE inspection standard that contains examples of acceptable and unacceptable visual indications and mockups for the VT-3 inspection of cracking. Input information needed includes:

- Identification of the most likely locations of surface irregularities, such as damaged or fractured material

- Identification of what visual examination indications are considered rejectable and would require additional examination and evaluation

Analytical efforts could be performed on a generic basis for all B&W-design units; however, the number of vent valves for DB is different from the rest of the units, which would require a unit-specific bypass analytical effort. The NDE inspection standard could also be developed generically.

Existing Documentation:

- CLB loadings (normal and faulted condition) are available, but a records search may need to be performed to identify them

What observations trigger examination into the Expansion category?

- Confirmed evidence of relevant conditions (damage, grossly cracked, or fractured material) in two or more vent valve discs shall require expansion to the CRGT spacer castings by the completion of the next refueling outage

Core Support Shield Vent Valve Retaining Rings and Disc Shaft

Item	Applicability	Effect (Mechanism)	Expansion Link	Examination Method/Frequency	Examination Coverage
Core Support Shield Assembly CSS vent valve top retaining ring CSS vent valve bottom retaining ring CSS vent valve disc shaft or hinge pin (Note 1)	All plants	Cracking (TE), including the detection of surface irregularities, such as damaged, fractured material, or missing items	None	Visual (VT-3) examination during the next 10-year ISI. Subsequent examinations on the 10-year ISI interval.	100% of accessible surfaces. (See BAW-2248A, page 4-3 and Table 4-1.) See MRP-231 Figures 3-10 and 3-11.

Notes:
1. A verification of the operation of each vent valve shall also be performed through manual actuation of the valve. Verify that the valves are not stuck in the open position and that no abnormal degradation has occurred. Examine the valves for evidence of scratches, pitting, embedded particles, variation in coloration of the seating surfaces, cracking of lock welds and locking cups, jack screws for proper position, and wear. The frequency is defined in each unit's technical specifications or in their pump and valve in-service test programs (see AREVA doc. BAW-2248A, page 4-3, and Table 4-1).

Vent valve top and bottom retaining rings and the disc shaft (or, hinge pin) are subject to thermal aging embrittlement, which if a flaw would be present and they are subjected to loading that exceeds the materials degraded fracture toughness, such a condition could potentially lead to cracking. Although there have been instances of failures of precipitation-hardenable materials in other applications, there is no known history of OE identifying cracking in PWR reactor vessel internals applications.

Component Item Function

Vent valves are passive devices that have no function during normal operation. The vent valve top and bottom retaining rings and the disc shaft (or, hinge pin) do not have a core support safety function; however, they do have a safety function in that degradation of the vent valve top and bottom retaining rings and the disc shaft (or, hinge pin), which would prevent the vent valve from opening, could result in loss of the vent valve function during a large break loss-of-coolant-accident (LOCA).

Observable Effects:

A visual (VT-3) examination of the accessible surfaces of the vent valve top and bottom retaining rings and the disc shaft (or, hinge pin) is to be performed. Subsequent visual examinations are to be performed during the ASME Code B-N-3 10-year ISI activities.

The vent valve top and bottom retaining rings and the disc shaft (or, hinge pin) are being examined to detect damage either in the form of a precursor to the loss of material, or, a piece or section of the material that has fractured or is currently missing, particularly, in the areas where high stresses exist. For example, with the retaining rings, at the locations where the jacking screws are connected.

Possible Examination Outcomes:

- No relevant conditions identified
- Observation that either retaining ring or disc shaft (or, hinge pin) is not in the correct position
- One or more areas are identified with crack-like indications
- One or more areas are identified with missing material

Methodology and Data Requirements:

The general analytical methodology to be used for acceptance criteria for the vent valve top and bottom retaining rings and the disc shaft (or, hinge pin) involves the following steps and inputs:

- Perform analysis to show that failure of vent valve items will not result in loss of function

- Perform a bypass analysis to justify that sufficient DNB exists in the degraded condition

- A VT-1, ET, or UT examination may be needed to determine if flaws are emanating from the location where missing material may be identified

The general methodology to be used for acceptance criteria for these component items will be development of an NDE inspection standard that contains examples of acceptable and unacceptable visual indications and mockups for the VT-3 inspection of cracking. Input information needed includes:

- Identification of the most likely locations of surface irregularities, such as damaged, fractured material, or missing items

- Identification of what visual examination indications are considered rejectable and would require additional examination and evaluation

Analytical efforts could be performed on a generic basis for all B&W units, although unit-specific analytical efforts may provide additional margin (particularly for DB). The NDE inspection standard could also be developed generically.

Existing Documentation:

- CLB loadings (normal and faulted condition) are available, but a records search may need to be performed to identify them

- Manufacturing and material data need to be identified to determine chemical composition and an assessment of the actual susceptibility to thermal aging embrittlement

- No acceptable evaluation or analysis has been completed to date for determining a re-inspection schedule

What observations trigger examination into the Expansion category?

There are no expansion items for these component items.

Upper Core Barrel Bolts and Locking Devices

Item	Applicability	Effect (Mechanism)	Expansion Link	Examination Method/Frequency	Examination Coverage
Core Support Shield Assembly Upper core barrel (UCB) bolts and their locking devices	All plants	Cracking (SCC)	LCB (Note 1) UTS, LTS, and FD bolts. SSHT bolts (CR-3 and DB only) Lower grid shock pad bolts (TMI-1 only)	Volumetric examination (UT) of the bolts within two refueling outages from 1/1/2006 or next 10-year ISI interval, whichever is first. Subsequent examination to be determined after evaluating the baseline results. Visual (VT-3) examination of bolt locking devices on the 10-year ISI interval.	100% of accessible bolts. See MRP-231 Figure 3-7.

Note:
1. Expansion to LCB applies if the required Primary examination of LCB has not been performed as scheduled in this table.

There is a potential for intergranular stress corrosion cracking (IGSCC) of Alloy A-286 and Alloy X-750 bolting. Past B&W failure history exists with the original Alloy A-286 bolt materials in B&W-design units and with applications of Alloy X-750 material within the nuclear industry (in general). Currently, there are no known failures with any of the replacement bolts (Alloy A-286 or Alloy X-750) in the operating B&W-design units or with the original Alloy X-750 (installed at TMI-1 only) in service in the operating B&W-design units.

Component Item Function

The UCB joint carries the entire weight of the core and the majority of the weight of the reactor vessel internals. The upper core barrel bolts have a core support safety function in that should the joint fail, the core and internals could drop, coming to rest on the guide lugs welded to the inside wall of the reactor vessel.

Observable Effects:

A volumetric examination (UT) of the bolts and a visual (VT-3) examination of the bolt locking devices is to be performed.

Mockups and qualification efforts exist for UCB and LCB bolts from the PWROG work (PA-MSC-350) and additional Duke Energy efforts in 2007-2008.

Cracking of the bolts is the main concern and the locking devices are to be examined to identify if any are distorted, loose, broken, or missing.

The PWROG work (PA-MSC-350) also evaluated the potential information that could be determined from only a visual examination of the bolt and locking devices (see AREVA document 51-9081184-001).

Possible Examination Outcomes:

Cracking is anticipated to occur at the head-to-shank area where the peak tensile stress exists (i.e., a SCF exists) and OE has shown them to crack at this location in the past, although it may also occur in the shank thread region where high tensile stress is possible too.

- No relevant conditions identified
- Relevant conditions (i.e., crack-like indications, either completely cracked or partially cracked; or non-interpretable UT indications, such as no back wall reflection or multiple reflections with no crack-like indication that is most likely caused by a large or duplex grain size) are identified
  - One or a few bolts (exact number is unit-specific) are identified with relevant indications
  - More than a few bolts (exact number is unit-specific) are identified with relevant indications

Locking Devices:

- No relevant conditions identified
- One or two are identified with damage or are missing
- More than two bolt locking devices are identified with damage or are missing

Methodology and Data Requirements:

The general analytical methodology to be used for acceptance criteria for the bolts involves the following steps and inputs if relevant conditions have been identified in the UCB bolts:

- A finite element model (FEM) is to be developed for the local geometry with contact conditions, pretension elements, loads and boundary conditions
- A thermal analysis is to be performed
  - Determines bolt temperatures and temperature gradients for normal operating conditions
- A structural analysis is to be performed in which failed bolts are inactive
  - Stress concentration factors are calculated to determine the peak stresses at the bolt head-to-shank fillet region under normal operating conditions

  - Analysis is performed for all loads and load combinations required for an ASME evaluation (stress limits for threaded structural fasteners in subsection NG and Appendix F)
  - The effect of failed or missing bolts on overall effective core barrel stiffness is evaluated
    - A change of no more than 20% in stiffness when subjected to LOCA loads is acceptable (within the limits of other uncertainties accounted for in the evaluation of LOCA loadings)
      - This corresponds to approximately 10% change in fundamental frequency of the structure
  - An evaluation of joint stability (or, openness) is also to be performed
- Representative rejected UCB bolts are to be removed for laboratory testing to confirm the UT inspection results and IGSCC mechanism, or to identify any other failure mechanism(s)

NOTE: One alternative to removing representative UT rejected bolts for laboratory examination is to re-inspect all bolts at the next refueling outage if continued operation for one additional fuel cycle can be supported by a technical evaluation. Other possible options such as replacement of a predefined bolt pattern may also be pursued.

- Based on the results of UCB bolt UT inspection and laboratory test results, perform an evaluation to assess future UCB bolt failure potential. The changes to the peak stress at the bolt head-to-shank fillet region as a result of the identified failures should be included for evaluation of increased susceptibility to SCC.
- Incorporate the effect of future UCB bolt failure into the operability evaluation and re-inspection requirement

The general methodology to be used for acceptance criteria for the locking devices will be development of an inspection standard that contains examples of acceptable and unacceptable locking device visual indications. The acceptance of locking devices is evaluated in two ways: a) observations with "failed" bolts and b) observations with all bolts intact. Observations of damaged locking devices with all bolts intact represent a condition very different from that of locking device damage at a bolt location that is failed. In addition, a damaged or missing welded locking clip versus a crimpled locking cup potentially represents different initiating phenomena that need to be evaluated.

Due to the variations in bolt materials used and loadings among the units, unit-specific analytical efforts are required. [The generic efforts have already been completed in the PWROG PA-MSC-350 work.] The NDE inspection standard could be developed generically.

<u>Existing Documentation:</u>

- A generic UCB bolt stress analysis was completed in the MRP reactor internals project for the three ONS units, CR-3, and ANO-1, which was subsequently used in the PA-MSC-0350 work noted below (see AREVA document 32-9095906).

- FEM and thermal analysis have been developed by the PWROG project (PA-MSC-0350, see AREVA document 51-9089393-000) to evaluate failures for use in evaluating an acceptable failure pattern or number of failed UCB and LCB bolts allowed for continued operation, and for use in preparing a JCO.

  – One or a few bolts could be identified as failed (non-interpretable bolt UT signal; inaccessible bolt for UT; locking device observed to be missing, non-functional, or removed; partially cracked bolt; or completely cracked bolt) and shown to be acceptable with no further action needed

  – A number of bolts (TBD) identified as failed (non-interpretable UT signal, partially cracked, or completely cracked) would initiate replacement activities and lead into the expansion category (see AREVA document 51-9087042-000 for initial methodology developed by PWROG project)

- Some unit-specific analyses for several assumed inspection cases have been performed for each unit in the PWROG project (PA-MSC-350), but unit-specific analyses may need to be re-performed for the actual inspection result (see AREVA document 51-9089393-000).

- If inspection results indicate no relevant indications of failure and calculated peak stresses are below the bolt material yield strength, SCC is not expected to initiate and an inspection during the next ASME Code B-N-3 10-year ISI interval is judged adequate

- If a relevant inspection condition is detected and confirmed by laboratory testing, a future bolt failure rate is needed for operability assessment and developing a future inspection frequency. No acceptable evaluation or analysis methodology has been developed to date for assessing future bolt failures or determining a re-inspection schedule based on the inspection results.

  – A suggested approach, based on a combinatoric risk analysis, has been provided to the PWROG (see PWROG MSC December 2008 meeting minutes) that is based on unit-specific inspection results and unit-specific bolt structural analysis.

An NRC-accepted crack growth rate for Alloy A-286 or Alloy X-750 material is not currently available. However, the PWROG project (PA-MSC-350, AREVA document 51-9079485-000) has identified some CGR data that is currently available. A CGR based life analysis has not been performed for any structural bolts. Based on the available CGR data, a life assessment based entirely on the CGR without considering crack initiation is judged unlikely to yield acceptable results.

What observations trigger examination into the Expansion category?

- Cracking observed in 10% (12) of the UCB bolts (LCB bolts are only considered in the expansion if they have not already been inspected). Damage to locking devices for failed bolt locations is not unusual and would be anticipated; however, if damage to locking devices is observed in non-failed bolt locations, the second trigger criterion would also be used.

- Observation of more than two locking devices damaged or missing (if no bolts are observed to be failed), pending additional evaluation as to the potential cause.

Should it trigger expansion to all remaining bolt rings or a tiered approach?

When an inspection triggers into the expansion, there is a unit-specific need to evaluate the results against the differences in materials used for the different locations and results from other unit inspections. For example, if failures are noted in the Alloy A-286 UCB bolts, but the UTS bolts are made from Alloy X-750 and no failures of this bolting material has been observed at any of the operating B&W units, a justification not to expand to this location should be possible. In addition, should failures be noted in a heat of Alloy A-286 used at one unit, expansion into the other units may need to be considered. However, one or more of the bolts with indications need to be removed for laboratory testing to confirm the IGSCC failure mechanism and stress analyses need to be performed for each of the expansion bolt locations. Thus, expansion is not to be considered carte-blanche without additional evaluation.

Lower Core Barrel Bolts and Locking Devices

Item	Applicability	Effect (Mechanism)	Expansion Link	Examination Method/Frequency	Examination Coverage
Core Barrel Assembly Lower core barrel (LCB) bolts and their locking devices	All plants	Cracking (SCC)	UTS, LTS, and FD bolts SSHT bolts (CR-3 and DB only) Lower grid shock pad bolts (TMI-1 only)	Volumetric examination (UT) of the bolts during the next 10-year ISI interval from 1/1/2006. Subsequent examination to be determined after evaluating the baseline results. Visual (VT-3) examination of bolt locking devices on the 10-year ISI interval.	100% of accessible bolts. See MRP-231 Figure 3-8.

There is a potential for intergranular stress corrosion cracking (IGSCC) of Alloy A-286 and Alloy X-750 bolting. Past B&W failure history exists with the original Alloy A-286 bolt materials in B&W-design units and with applications of Alloy X-750 material within the nuclear industry (in general). Currently, there are no known failures with any of the replacement bolts (Alloy A-286 or Alloy X-750) in the operating B&W-design units or with the original Alloy X-750 (installed at TMI-1 only) in service in the operating B&W-design units.

Component Item Function

The LCB joint carries the entire weight of the core (but not the weight of the core barrel) and the weight of the lower reactor vessel internals. The lower core barrel bolts have a core support safety function in that should the joint fail, the core and lower internals could drop, coming to rest on the guide lugs welded to the inside wall of the reactor vessel.

Observable Effects:

A volumetric examination (UT) of the bolts and a visual (VT-3) examination of the bolt locking devices

Mockups and qualification efforts exist for UCB and LCB bolts from the PWROG work (PA-MSC-350) and additional Duke Energy efforts in 2007-2008.

Cracking of the bolts is the main concern and the locking devices are to be examined to identify if any are distorted, loose, broken, or missing.

The PWROG work (PA-MSC-350) also evaluated the potential information that could be determined from only a visual examination of the bolt and locking devices (see AREVA document 51-9081184-001).

Possible Examination Outcomes:

Bolts

Cracking is anticipated to occur at the head-to-shank area where the peak tensile stress exists (i.e., a SCF exists) and OE has shown them to crack at this location in the past, although it may also occur in the shank thread region where high tensile stress is possible too.

- No relevant conditions identified

- Relevant conditions (i.e., crack-like indications, either completely cracked or partially cracked; or non-interpretable UT indications, such as no back wall reflection or multiple reflections with no crack-like indication that is most likely caused by a large or duplex grain size) are identified

  - One or a few bolts (exact number is unit-specific) are identified with relevant indications

  - More than a few bolts (exact number is unit-specific) are identified with relevant indications

Locking Devices

- No relevant conditions identified
- One or two are identified with damage or are missing
- More than two bolt locking devices are identified with damage or are missing

Methodology and Data Requirements:

The general analytical methodology to be used for acceptance criteria for the bolts involves the following steps and inputs if relevant conditions have been identified in the LCB bolts:

- A finite element model (FEM) is to be developed for the local geometry with contact conditions, pretension elements, loads and boundary conditions

- A thermal analysis is to be performed

  - Determines bolt temperatures and temperature gradients for normal operating conditions

- A structural analysis is to be performed in which failed bolts are inactive

  - Stress concentration factors are calculated to determine the peak stresses at the bolt head-to-shank fillet region under normal operating conditions

  - Analysis is performed for all loads and load combinations required for an ASME evaluation (stress limits for threaded structural fasteners in subsection NG and Appendix F)

  - An evaluation of joint stability (or, openness) is also to be performed

- Representative rejected LCB bolts are to be removed for laboratory testing to confirm the UT inspection results and IGSCC mechanism, or to identify any other failure mechanism(s)

NOTE: One alternative to removing representative UT rejected bolts for laboratory examination is to re-inspect all bolts at the next refueling outage if continued operation for one additional fuel cycle can be supported by a technical evaluation. Other possible options such as replacement of a predefined bolt pattern may also be pursued.

- Based on the results of LCB bolt UT inspection and laboratory test results, perform an evaluation to assess future LCB bolt failure potential. The changes to the peak stress at the bolt head-to-shank fillet region as a result of the identified failures should be included for evaluation of increased susceptibility to SCC.

- Incorporate the effect of future LCB bolt failure into the operability evaluation and re-inspection requirement

The general methodology to be used for acceptance criteria for the locking devices will be development of an inspection standard that contains examples of acceptable and unacceptable locking device visual indications. The acceptance of locking devices is evaluated in two ways: a) observations with "failed" bolts and b) observations with all bolts intact. Observations of damaged locking devices with all bolts intact represent a condition very different from that of locking device damage at a bolt location that is failed. In addition, a damaged or missing welded locking clip versus a crimpled locking cup potentially represents different initiating phenomena that need to be evaluated.

Due to the variations in bolt materials used and loadings among the units, unit-specific analytical efforts are required. [The generic efforts have already been completed in the PWROG PA-MSC-350 work.] The NDE inspection standard could be developed generically.

<u>Existing Documentation:</u>

- A generic LCB bolt stress analysis was completed in the MRP reactor internals project for the three ONS units and ANO-1, which was subsequently used in the PA-MSC-0350 work noted below (see AREVA document 32-9095906).

- FEM and thermal analysis have been developed by the PWROG project (PA-MSC-0350, see AREVA document 51-9089393-000) to evaluate failures for use in evaluating an acceptable failure pattern or number of failed UCB and LCB bolts allowed for continued operation, and for use in preparing a JCO.

  – One or a few bolts could be identified as failed (non-interpretable bolt UT signal; inaccessible bolt for UT; locking device observed to be missing, non-functional, or removed; partially cracked bolt; or completely cracked bolt) and shown to be acceptable with no further action needed

  – A number of bolts (TBD) identified as failed (non-interpretable UT signal, partially cracked, or completely cracked) would initiate replacement activities and lead into the expansion category (see AREVA document 51-9087042-000 for initial methodology developed by PWROG project)

- Some unit-specific analyses for several assumed inspection cases have been performed for each unit in the PWROG project (PA-MSC-350), but unit-specific analyses may need to be re-performed for the actual inspection result (see AREVA document 51-9089393-000).

- If inspection results indicate no relevant indications of failure and calculated peak stresses are below the bolt material yield strength, SCC is not expected to initiate and an inspection during the next ASME Code B-N-3 10-year ISI interval is judged adequate

- If a relevant inspection condition is detected and confirmed by laboratory testing, a future bolt failure rate is needed for operability assessment and developing a future inspection frequency. No acceptable evaluation or analysis methodology has been developed to date for assessing future bolt failures or determining a re-inspection schedule based on the inspection results.

  – A suggested approach, based on a combinatoric risk analysis, has been provided to the PWROG (see PWROG MSC December 2008 meeting minutes) that is based on unit-specific inspection results and unit-specific bolt structural analysis.

An NRC-accepted crack growth rate for Alloy A-286 or Alloy X-750 material is not currently available. However, the PWROG project (PA-MSC-350, AREVA document 51-9079485-000) has identified some CGR data that is currently available. A CGR based life analysis has not been performed for any structural bolts. Based on the available CGR data, a life assessment based entirely on the CGR without considering crack initiation is judged unlikely to yield acceptable results.

<u>What observations trigger examination into the Expansion category?</u>

- Cracking observed in 10% (11) of the LCB bolts. Damage to locking devices for failed bolt locations is not unusual and would be anticipated; however, if damage to locking devices is observed in non-failed bolt locations, the second trigger criterion would also be used.

- Observation of more than two locking devices damaged or missing (if no bolts are observed to be failed), pending additional evaluation as to the potential cause.

<u>Should it trigger expansion to all remaining bolt rings or a tiered approach?</u>

When an inspection triggers into the expansion, there is a unit-specific need to evaluate the results against the differences in materials used for the different locations and results from other unit inspections. For example, if failures are noted in the Alloy A-286 LCB bolts, but the UTS bolts are made from Alloy X-750 and no failures of this bolting material has been observed at any of the operating B&W units, a justification not to expand to this location should be possible. In addition, should failures be noted in a heat of Alloy A-286 used at one unit, expansion into the other units may need to be considered. However, one or more of the bolts with indications need to be removed for laboratory testing to confirm the IGSCC failure mechanism and stress analyses need to be performed for each of the expansion bolt locations. Thus, expansion is not to be considered carte-blanche without additional evaluation.

Baffle-to-Former Bolts

Item	Applicability	Effect (Mechanism)	Expansion Link	Examination Method/Frequency	Examination Coverage
Core Barrel Assembly Baffle-to-former bolts	All plants	Cracking (IASCC, IE, IC/ISR/ Fatigue/Wear, Overload)	Baffle-to-baffle bolts Core barrel-to-forme r bolts	Baseline volumetric examination (UT) no later than two refueling outages from the beginning of the license renewal period with subsequent examination after 10 to 15 additional years.	100% of accessible bolts. See MRP-231 Figure 3-2.

There is a potential for failure in the form of cracking of baffle-to-former bolting to occur mainly from irradiation-assisted stress corrosion cracking (IASCC), but also as a result of irradiation embrittlement, irradiation creep/stress relaxation (leading to fatigue and wear), or overload (from a prying effect). Past failure history exists with baffle-to-former bolt materials (Type 316CW and Type 347) and core barrel-to-former bolt materials (Alloy X-750 and Type 316Ti CW) in non-B&W-design units. Currently, there are no known failures with any of the bolts (Type 304) in the operating B&W-design units. CR-3 has observed what appear to possibly be failed baffle-to-baffle bolts, but confirmation has never been made.

Component Item Function

The core barrel assembly consists of the core barrel cylinder, former plates, and baffle plates connected by bolted joints that include: (1) core barrel-to-former bolts (CF bolts), (2) baffle-to-former bolts (BF bolts), and (3) baffle-to-baffle bolts (BB bolts). The core barrel assembly supports the fuel assemblies, lower grid, flow distributor, and incore instrument guide tubes. The baffle plates, former plates, and their joints (including BF bolts) do not have a core support function and are categorized as internals structures. The primary function of the baffle plates, former plates, and their bolted connections is to provide a flow envelope surrounding the core. Also, since they are bolted to the core barrel cylinder, the baffle plates and former plates will produce a small increase on the stiffness and natural frequencies of the core barrel assembly.

The CF and BF bolts have the function of maintaining structural integrity of the baffle and former portion of the structural assembly and thus of maintaining flow geometry during normal operation. For faulted events, a small number of the CF and BF bolts are needed to restrain the baffle so that a coolable core geometry is maintained.

As with BB and CF bolts, loss of BF bolts will also influence changes in the core bypass flow due to opening of baffle-to-baffle corner gaps.

Observable Effects:

Cracking of the bolts is the main concern. A volumetric examination (UT) of the bolts is to be performed to identify such cracking.

Mockups and qualification efforts exist from the CR-3 examination in 2005, but there are several bolt designs in the B&W units and additional effort would still need to be completed.

Possible Examination Outcomes:

Cracking is anticipated to occur at the head-to-shank area where the peak tensile stress exists (i.e., a SCF exists) and OE has shown them to crack at this location in the past, although it may also occur in the shank thread region where high tensile stress is possible too.

- No relevant conditions identified
- Relevant conditions (i.e., crack-like indications, either completely cracked or partially cracked; or non-interpretable UT indications, such as no back wall reflection or multiple reflections with no crack-like indication that is most likely caused by a large or duplex grain size) are identified

Methodology and Data Requirements:

The general analytical methodology to be used for acceptance criteria for the BF bolts involves the following steps and inputs:

- A global finite element model (FEM) is developed to evaluate bolt failures for use in developing the frequency for the I&E guidelines, acceptable failure pattern or numbers, and for use in preparing possible JCOs for the BF bolts
- Representative rejected BF bolts are to be removed for laboratory testing to confirm the UT inspection results and IGSCC mechanism, or to identify any other failure mechanism(s)

  NOTE: One alternative to removing representative UT rejected bolts for laboratory examination is to re-inspect all bolts at the next refueling outage if continued operation for one additional fuel cycle can be supported by a technical evaluation. Other possible options such as replacement of a predefined bolt pattern may also be pursued.

- The following inputs are required:
  - Failed or missing BF bolt locations are required for input
  - Thermal input including gamma-heating for design (short-term) and long-term operating conditions

- Irradiated material property input as a function of aging (EFPY) of the core barrel assembly
- Applicable test data to establish stress and/or strain and fatigue strength limits of the BF bolts in addition to the licensing basis requirements
- Faulted load licensing basis requirements based on existing evaluations and modified as needed
- Input of existing fuel baffle jetting evaluations and their applicability for the core baffle assembly in degraded conditions and modified as needed to establish if gap displacements are necessary acceptance criteria
- Input core barrel motion due to turbulence induced vibration from applicable startup testing and analytical evaluations

- A probability of failure of the inaccessible core barrel-to-former bolts and external baffle-to-baffle bolts will need to be determined
- Appropriate structural evaluations are performed to demonstrate the above acceptance criteria
- If necessary, the existing model will be modified to be suitable for dynamic loadings such as imposed core barrel motion due to turbulence induced vibration

Analytical efforts could be performed on a generic basis, although there are two designs (ONS-1 and TMI-1 are one design and the other five units are the second design).

Existing Documentation:

- A FEM has been developed by the MRP Reactor Internals project that can be used in performing the evaluations
  - A few bolts could be identified as failed (non-interpretable bolt UT signal; inaccessible bolt for UT; locking device observed to be missing, non-functional, or removed; partially cracked bolt; or completely cracked bolt) and shown to be acceptable with no further action needed
  - A number of bolts (TBD) identified as failed (non-interpretable UT signal, partially cracked, or completely cracked) would initiate replacement activities and lead into the expansion category
    - Past B&WOG work has determined the minimum number of bolts required for safe shutdown, but not for operation; i.e., no minimum bolt patterns have been determined

- No acceptable evaluation or analysis has been completed to date for determining a re-inspection schedule

<u>What observations trigger examination into the Expansion category?</u>

- Cracking observed in 5% (40) of the bolts or greater than 25% of the bolts on a single former plate

<u>Should it trigger expansion to all remaining bolt rings or a tiered approach?</u>

When an inspection triggers into the expansion, an evaluation of the internal BB bolts shall be performed to determine whether to examine or replace the internal BB bolts. The evaluation may also include an evaluation of the external BB bolts and CF bolts for the purpose of determining whether to continue operation or further expand into replacement activities.

Baffle Plates

Item	Applicability	Effect (Mechanism)	Expansion Link	Examination Method/Frequency	Examination Coverage
Core Barrel Assembly Baffle plates	All plants	Cracking (IE), including the detection of readily detectable cracking in the baffle plates	Core barrel cylinder (including vertical and circumferential seam welds) Former plates	Visual (VT-3) examination during the next 10-year ISI. Subsequent examinations on the 10-year ISI interval.	100% of the accessible surface within 1 inch around each flow and bolt hole. See MRP-231 Figure 3-2.

Baffle plates are subject to irradiation embrittlement, which if a flaw would be present and they are subjected to loading that exceeds the materials degraded fracture toughness, such a condition could potentially lead to cracking. There is no known history of cracking of baffle plate material in PWR reactor vessel internals applications.

Component Item Function

Degradation of the baffle plates could result in increased core bypass flow and a reduction in margin to DNB, but would probably have a negligible effect on unit operations and would not be observed except by direct examination. The core barrel supports the fuel assemblies, lower grid, flow distributor, and incore instrument guide tubes. However, the baffle plates do not support any load. The primary function of the baffle plates during normal power operation is to provide a flow envelope for the core and, thereby limit core bypass flow.

The baffle plates therefore do not have a direct core support safety function; however, they do have a safety function to control bypass around the core during a loss-of-coolant-accident (LOCA).

Observable Effects:

A visual (VT-3) examination of the baffle plates is to be performed. Subsequent visual examinations are to be performed during the ASME Code B-N-3 10-year ISI activities.

If flaws are identified, follow-on examination by VT-1, ET, or UT may need to be performed to characterize the length or both length and depth of observations.

The baffle plates are being examined to detect large surface cracks. The locations expected to be subjected to the highest tensile stresses are near the baffle bolt holes (baffle-to-former and baffle-to-baffle) and flow holes/slots. Examinations should also include areas near the HAZ of the baffle bolt locking devices where residual tensile stresses may exist.

Possible Examination Outcomes:

- No relevant conditions identified
- One or more areas are identified with minor (short) crack-like indications

- One or more areas are identified with large (long) crack-like indications
- One or more areas are identified with missing material

Methodology and Data Requirements:

The general analytical methodology to be used for acceptance criteria for the baffle plates involves the following steps and inputs:

- Confirmation of required loading and combination requirements

- Determine the expected crack opening displacement (COD) for development of the inspection standard

- Perform a linear-elastic fracture mechanics (LEFM) evaluation to determine the critical crack size using the MRP-211 fracture toughness values

  - A flaw handbook could also be developed
  - Or, justify the existing calculations in MRP-210

- Perform a bypass analysis to justify that sufficient DNB exists in the degraded condition

- A VT-1, ET, or UT examination may be needed to further characterize the flaws or to determine if flaws are emanating from the location where missing material may be identified

- An operability evaluation to operate at least one cycle based on possible inspection results for the plates should be performed

- An evaluation of the consequences of leaving cracked plates securely in place during an inspection or replacement campaign should be performed

The general methodology to be used for acceptance criteria for the baffle plates will be development of an NDE inspection standard that contains examples of acceptable and unacceptable visual indications and mockups for the VT-3 inspection of cracking. Input information needed includes:

- Identification of the most likely locations of cracking in the plates

- Identification of what visual examination indications are considered rejectable and would require additional examination and evaluation

Analytical efforts could be performed on a generic basis. The NDE inspection standard could also be developed generically.

Existing Documentation:

- CLB loadings (normal and faulted condition) are available, but a records search may need to be performed to identify them
- No CGRs currently exist
  - CGRs for BWR HWC can be assumed for initial studies

What observations trigger examination into the Expansion category?

- Gross cracking (if confirmed) on two or more locations in the baffle plates shall trigger an evaluation of the inspectability of the accessible areas of the former plates and core barrel (particularly the core barrel upper flange-to-core barrel weld and upper HAZ area) using VT-3 inspection
- In addition, an evaluation of the operability of the former plates and core barrel in degraded conditions shall be performed
- If a VT-3 examination is possible, it is required by completion of the next refueling outage

Locking Devices of Baffle-to-Former and Internal Baffle-to-Baffle Bolts

Item	Applicability	Effect (Mechanism)	Expansion Link	Examination Method/ Frequency	Examination Coverage
Core Barrel Assembly Locking devices, including locking welds, of baffle-to-former bolts and internal baffle-to-baffle bolts	All plants	Cracking (IASCC, IE, Overload), including the detection of missing, non-functional, or removed locking devices or welds	Locking devices for the external baffle-to-baffle bolts and core barrel-to-former bolts	Visual (VT-3) examination during the next 10-year ISI. Subsequent examinations on the 10-year ISI interval.	100% of accessible baffle-to-former and internal baffle-to-baffle bolt locking devices. See MRP-231 Figure 3-2.

There is a potential for irradiation-assisted stress corrosion cracking (IASCC) of the locking devices or welds for the baffle (baffle-to-former and internal baffle-to-baffle) bolting. There is also the potential for failure by overload for these locking devices and welds due to slip between the bolts and plates (see MRP-231). Past failure history exists with cracked and missing internal baffle-to-baffle bolt locking devices and cracked locking device welds in B&W-design units.

Component Item Function

The locking devices and welds are not normally loaded unless the bolt is broken or loose. Loading of the locking devices and welds could also occur due to the slip between the bolt and plate. The locking devices and welds have no core support safety function.

Observable Effects:

A visual (VT-3) examination of the bolt locking devices and welds is to be performed.

Cracking of the locking devices or welds is the main concern and they are to be examined to identify if any are distorted, loose, broken, or missing.

Possible Examination Outcomes:

- No relevant conditions identified
- One or a few bolt locking devices (up to 1% or 11) are identified with damage or are missing
- More than 1% (or, 11) bolt locking devices are identified with damage or are missing

Methodology and Data Requirements:

The general methodology to be used for acceptance criteria for the locking devices and welds will be development of an NDE inspection standard that contains examples of acceptable and unacceptable locking device or weld visual indications. The acceptance of locking devices and welds is evaluated in two ways: a) observations with "failed" bolts and b) observations with all bolts intact. Observations of damaged locking devices or welds with all bolts intact represent a condition very different from that of locking device or weld damage at a bolt location that is failed. In addition, a damaged or missing welded locking washer versus a welded locking pin or bar potentially represents different initiating phenomena that need to be evaluated.

If bolt failure is not obvious, additional UT examinations of the bolt or a technical justification for removal of the locking device and bolt may be necessary.

The NDE inspection standard for the locking devices and welds could be performed on a generic basis for the B&W units, although there are two designs (ONS-1 and TMI-1 are one design and the other five units are the second design).

Existing Documentation:

Damaged and missing locking devices and welds from the CR-3 visual examinations can be used for development of an inspection standard, although additional efforts for other locking devices and welds will also be required.

What observations trigger examination into the Expansion category?

- Confirmed rejectable indications in greater than or equal to 1% (or, 11) of the baffle-to-former and internal BB bolt locking devices shall trigger an evaluation of the locking devices for the external baffle-to-baffle bolts and core barrel-to-former bolts for the purpose of determining continued operation or replacement

Guide Block Dowel Welds

Item	Applicability	Effect (Mechanism)	Expansion Link	Examination Method/Frequency	Examination Coverage
Lower Grid Assembly Alloy X-750 dowel-to-guide block welds	All plants	Cracking (SCC), including the detection of separated or missing locking welds, or missing dowels	Alloy X-750 dowel locking welds to the upper and lower grid fuel assembly support pads	Initial visual (VT-3) examination no later than two refueling outages from the beginning of the license renewal period. Subsequent examinations on the 10-year ISI interval.	100% of accessible locking welds of the 24 dowel-to-guide block welds. See MRP-231 Figure 3-4.

There is a potential for primary water stress corrosion cracking (PWSCC) of the dowel-to-guide block welds. No known OE exists for cracking in these welds either in B&W-design units or other vendor designs, but OE with cracking of Alloy 82/182 weld materials in PWRs does exist.

Component Item Function

These welds serve as loose part prevention devices and are not structural. Small cracks in the locking weld are acceptable since the locking function can be maintained as long as any part of the weld is present.

The dowel-to-guide block welds have no core support safety function.

Observable Effects:

A visual (VT-3) examination of the dowel-to-guide block welds is to be performed.

Loss of the locking function of the weld is the main concern and therefore the dowel-to-guide block welds are to be examined to identify if any are separated or missing or if a dowel is missing.

Possible Examination Outcomes:

- No relevant conditions exist
- A single weld is observed to be damaged or partially missing
- Several welds are observed to be damaged or partially missing
- A single weld or dowel is missing
- Several welds or dowels are missing
- A guide block is misaligned or is missing
- Several guide blocks are misaligned or are missing

Methodology and Data Requirements:

The general methodology to be used for acceptance criteria for the dowel-to-guide block welds will be development of an NDE inspection standard that contains examples of acceptable and unacceptable dowel-to-guide block weld visual indications. The function of the weld can be maintained as long as a portion of the weld is in place. Significant cracking of the weld and subsequent loss of the dowel does not compromise the function of the guide block unless the bolt also fails.

The following items will be examined to establish VT-3 acceptance criteria and the technical justification:

- Identify normal and faulted operating loads for the guide block dowels
- Evaluate how many (if any) guide blocks are needed for operation
- Evaluate the consequences of leaving partially cracked locking welds securely in place during an inspection
- Identify the areas to be examined containing what is rejectable and requiring further evaluation
- Develop repair strategies for leaving in place if secured from being a loose part or removal and replacement activities

A UT examination of the guide block bolt or a technical justification for removal of the guide block may be necessary.

Analytical efforts could be performed on a generic basis for the B&W units. The NDE inspection standard could also be developed generically.

Existing Documentation:

Minimal information is currently available.

What observations trigger examination into the Expansion category?

- Confirmed rejectable indications of two or more separated, cracked, or failed locking welds shall trigger a VT-3 examination of the expansion category locking welds by the next scheduled refueling outage

Incore Monitoring Instrumentation Guide Tube Spiders and Welds

Item	Applicability	Effect (Mechanism)	Expansion Link	Examination Method/Frequency	Examination Coverage
Incore Monitoring Instrumentation (IMI) Guide Tube Assembly IMI guide tube spiders IMI guide tube spider-to-lower grid rib section welds	All plants	Cracking (TE/IE), including the detection of fractured or missing spider arms or separation of spider arms from the lower grid rib section at the weld	CRGT spacer castings Lower grid fuel assembly support pad items: pad, pad-to-rib section welds, Alloy X-750 dowel, cap screw, and their locking welds (Note: The pads, dowels, and cap screws are included because of TE/IE of the welds)	Initial visual (VT-3) examination no later than two refueling outages from the beginning of the license renewal period. Subsequent examinations on the 10-year interval.	100% of accessible top surfaces of 52 spider castings and welds to the adjacent lower grid rib section. See MRP-231 Figures 3-3 and 3-6.

The IMI guide tube spiders and welds are subject to both thermal aging and irradiation embrittlement, which if a flaw would be present and they are subjected to loading that exceeds the materials degraded fracture toughness, such a condition could potentially lead to cracking. There is currently no known history of cracking of CASS or weld material subjected to such embrittlement in PWR reactor vessel internals applications.

Component Item Function

Degradation of the IMI guide tube spiders could result in misalignment of the IMI nozzles and subsequent insertion of the in-core monitoring instrumentation. The IMI guide tube spiders do not have a core support safety function.

Observable Effects

A visual (VT-3) examination of the IMI guide tube spiders is to be performed. Subsequent visual examinations are to be performed during the ASME Code B-N-3 10-year ISI activities.

The IMI guide tube spiders are being examined to detect spider arms that do not align with the lower fuel assembly support pad center bolt. The location that potentially contains the highest tensile stresses is near the heat-affected-zone (HAZ) of the spider-to-lower grid rib section weld, which is not readily accessible.

Possible Examination Outcomes:

- No relevant conditions identified
- One or more areas are identified with a spider arm that is not aligned with the lower fuel assembly support pad center bolt or obvious separation or the spider arm from the lower grid rib section welded connection
- One or more areas are identified with a missing spider arm

Methodology and Data Requirements:

The general analytical methodology to be used for acceptance criteria for the IMI guide tube spiders involves the following steps and inputs:

- Prepare an analysis to show that one or more missing spider arms or a completely missing spider will not result in loss of function of the IMI guide tube
- A VT-1, ET, or UT examination may be needed to determine if flaws are emanating from the location where missing material may be identified

The general methodology to be used for acceptance criteria for these component items will be development of an NDE inspection standard that contains examples of acceptable and unacceptable visual indications and mockups for the VT-3 inspection of cracking. Input information needed includes:

- Identification of the most likely locations of surface irregularities, such as fractured or missing spider arms or separation of spider arms from the lower grid rib section at the weld
- Identification of what visual examination indications are considered rejectable and would require additional examination and evaluation

Analytical efforts could be performed on a generic basis for each of the B&W units. The NDE inspection standard could also be developed generically.

Existing Documentation:

- CLB loadings (normal and faulted condition) are available, but a records search may need to be performed to identify them

What observations trigger examination into the Expansion category?

Confirmed evidence of misalignment, separation, gross damage, or a missing spider arm for two or more locations shall trigger VT-3 examination of 100% of the accessible surfaces at the 4 screw locations (at every 90°) of the CRGT spacer castings and the lower grid assembly support pad items and it is required by completion of the next refueling outage.

Expansion Component Items

Acceptance criteria methodology and data requirements for each of the Expansion component items are summarized in this section. A separate sub-section is provided for each component item using the following format:

- Expansion component item information extracted directly from Table 4-4 of MRP-227
- This information is in tabular form and contains the item name, unit applicability, failure effect, failure mechanism(s), expansion link(s), examination method, examination frequency, and examination coverage.
- Component item function(s), including whether or not it has a core support safety function
- Observable effect(s)
- Methodology for development of acceptance criteria
- Data requirements for development of acceptance criteria
- Existing documents (e.g., PWROG or AREVA)

Upper and Lower Grid Fuel Assembly Support Pad Dowel Welds

Item	Applicability	Effect (Mechanism)	Primary Link	Examination Method	Examination Coverage
Upper or Lower Grid Assembly Alloy X-750 dowel-to-upper grid fuel assembly support pad welds or Alloy X-750 dowel-to-lower grid fuel assembly support pad welds	All plants (except upper grid assembly at DB)	Cracking (SCC), including the detection of separated or missing locking weld, or missing dowels	Alloy X-750 dowel-to-guide block welds	Visual (VT-3) examination.	100% of accessible dowel locking welds. See MRP-231 Figure 3-6.

There is a potential for primary water stress corrosion cracking (PWSCC) of the dowel-to-grid fuel assembly support pad (both upper and lower grids) welds. No known OE of cracking exists for these welds either in B&W-design units or other vendor designs, but OE with cracking of Alloy 82/182 weld materials in PWRs does exist.

Component Item Function

These welds serve as loose part prevention devices and are not structural. Small cracks in the locking weld are acceptable since the locking function can be maintained as long as any part of the weld is present. The fuel assembly support pads serve as guidance for loading of the fuel into the core. Once the fuel assemblies are loaded into the core, the support pads no longer have any function.

Therefore, the dowel-to-grid fuel assembly support pad welds have no core support safety function.

Observable Effects:

A visual (VT-3) examination of the dowel-to-grid fuel assembly support pad (upper and lower grids) welds is to be performed.

Loss of the locking function of the weld is the main concern and therefore the dowel-to-grid fuel assembly support pad welds are to be examined to identify if any are separated or missing, if a dowel is missing, or the support pad is misaligned (clearly out of perpendicularity).

Possible Examination Outcomes:

- No relevant conditions exist
- A single weld is observed to be damaged or partially missing
- Several welds are observed to be damaged or partially missing
- A single weld or dowel is missing
- Several welds or dowels are missing
- A support pad is misaligned (clearly out of perpendicularity) or missing
- Several support pads are misaligned (clearly out of perpendicularity) or missing

Methodology and Data Requirements:

The general methodology to be used for acceptance criteria for the dowel-to-grid fuel assembly support welds will be development of an NDE inspection standard that contains examples of acceptable and unacceptable dowel-to-grid fuel assembly support pad welds visual indications. The function of the weld can be maintained as long as a portion of the weld is in place. Significant cracking of the weld and subsequent loss of the dowel does not compromise the function of the fuel assembly support pad unless the bolt also fails.

The following items will be examined to establish VT-3 acceptance criteria and the technical justification:

- Identify normal and faulted operating loads for the fuel assembly support pad dowels
- Evaluate the consequences of leaving partially cracked locking welds securely in place during an inspection
- Identify the areas to be examined containing what is rejectable and requiring further evaluation
- Develop repair strategies for leaving in place if secured from being a loose part or removal and replacement activities

A UT examination of the fuel assembly support pad bolt or a technical justification for removal of the fuel assembly support pad may be necessary.

Analytical efforts could be performed on a generic basis for the applicable locations at each of the B&W units. The NDE inspection standard could also be developed generically.

Existing Documentation:

Minimal information is currently available.

Control Rod Guide Tube Spacer Castings

Item	Applicability	Effect (Mechanism)	Primary Link	Examination Method	Examination Coverage
Control Rod Guide Tube Assembly CRGT spacer castings	All plants	Cracking (TE), including the detection of fractured spacers or missing screws	CSS cast outlet nozzle CSS vent valve discs IMI guide tube spiders	Visual (VT-3) examination.	100% of accessible surfaces at the 4 screw locations (at every 90°) (Limited accessibility). See MRP-231 Figure 3-5.

CRGT spacer castings are subject to thermal aging embrittlement, which if a flaw would be present and they are subjected to loading that exceeds the materials degraded fracture toughness, such a condition could potentially lead to cracking. There is no known history of cracking of CASS material in PWR reactor vessel internals applications.

Component Item Function

Degradation of the spacer castings could result in degradation in the unit shutdown capability. The spacer castings do not have a core support safety function; however, they do have a safety function relative to control rod alignment, insertion and reactivity issues, and a stuck rod scenario.

Observable Effects

A visual (VT-3) examination of the CRGT spacer castings is to be performed.

The spacer castings have limited accessibility from the top or bottom of the CRGT through a center free-path (once the plenum assembly is removed from the vessel). Examination at the quarter points where the threaded connections are present is possible. These lanes are not blocked by the rod guide tubes. The examination would look for cracking of the spacer surface or evidence that the spacer is not approximately centered. The threaded fasteners are welded to the OD of the pipe column so it is possible that a degraded threaded location would not be detected.

Possible Examination Outcomes:

- No relevant conditions identified
- One or more areas are identified with a large crack like indication that would be a precursor to loosing a piece of material
- One or more areas are identified with missing material
- Evidence that the spacer is not centered

Methodology and Data Requirements:

The general analytical methodology to be used for acceptance criteria for the CRGT spacer castings involves the following step and input:

- Perform a reactivity analysis to determine the number of CRDMs that are required for shut down of the reactor

- A VT-1, ET, or UT examination may be needed to determine if flaws are emanating from the location where missing material may be identified.

The general methodology to be used for acceptance criteria for these component items will be development of an NDE inspection standard that contains examples of acceptable and unacceptable visual indications and mockups for the VT-3 inspection of fractured spacers or missing screws.

Analytical efforts for reactivity analyses are dependent upon fuel loading and must be performed on a unit-specific basis. The NDE inspection standard could also be developed generically.

Existing Documentation:

- CLB loadings (normal and faulted condition) are not currently available or even possibly required for analyses, but could be easily developed

- Core design

Upper Thermal Shield Bolts and Locking Devices

Item	Applicability	Effect (Mechanism)	Primary Link	Examination Method	Examination Coverage
Core Barrel Assembly Upper thermal shield bolts (UTS) and their locking devices	All plants	Cracking (SCC)	UCB bolts LCB bolts	Volumetric examination (UT). Visual (VT-3) examination of bolt locking devices on the 10-year ISI interval.	100% of accessible bolts. See MRP-231 Figure 3-7.

There is a potential for intergranular stress corrosion cracking (IGSCC) of Alloy A-286 and Alloy X-750 bolting. Past B&W failure history exists with the original Alloy A-286 bolt materials in B&W-design units and with applications of Alloy X-750 material within the nuclear industry (in general). Currently, there are no known failures with any of the replacement bolts (Alloy A-286 or Alloy X-750) in the operating B&W-design units or with the original Alloy X-750 (installed at TMI-1 only) in service in the operating B&W-design units.

Component Item Function

The UTS bolts fasten a split restraint and shim block to the core barrel. The UTS bolts do not have a core support safety function.

Observable Effects:

A volumetric examination (UT) of the bolts and a visual (VT-3) examination of the bolt locking devices

Mockups and qualification efforts exist from the PWROG work (PA-MSC-350) and additional Duke Energy efforts in 2007-2008.

Cracking of the bolts is the main concern and the locking devices are to be examined to identify if any are distorted, loose, broken, or missing.

The PWROG work (PA-MSC-350) also evaluated the potential information that could be determined from only a visual examination of the bolt and locking devices (see AREVA document 51-9081184-001).

Possible Examination Outcomes:

Cracking is anticipated to occur at the head-to-shank area where the peak tensile stress exists (i.e., a SCF exists) and OE has shown them to crack at this location in the past, although it may also occur in the shank thread region where high tensile stress is possible too.

- No relevant conditions identified

- Relevant conditions (i.e., crack-like indications, either completely cracked or partially cracked; or non-interpretable UT indications, such as no back wall reflection or multiple reflections with no crack-like indication that is most likely caused by a large or duplex grain size) are identified

  - One or a few bolts (exact number is unit-specific) are identified with relevant indications

  - More than a few bolts (exact number is unit-specific) are identified with relevant indications

Locking Devices

- No relevant conditions identified
- One or two are identified with damage or are missing
- More than two bolt locking devices are identified with damage or are missing

Methodology and Data Requirements:

The general analytical methodology to be used for acceptance criteria for the bolts involves the following steps and inputs if relevant conditions have been identified in the UTS bolts:

- A finite element model (FEM) is to be developed for the local geometry with contact conditions, pretension elements, loads and boundary conditions

- A thermal analysis is to be performed

  - Determines bolt temperatures and temperature gradients for normal operating conditions

- A structural analysis is to be performed in which failed bolts are inactive

  - Stress concentration factors are calculated to determine the peak stresses at the bolt head-to-shank fillet region under normal operating conditions

  - Analysis is performed for all loads and load combinations required for an ASME evaluation (stress limits for threaded structural fasteners in subsection NG and Appendix F)

  - An evaluation of joint stability (or, openness) is also to be performed

- Representative rejected UTS bolts are to be removed for laboratory testing to confirm the UT inspection results and IGSCC mechanism, or to identify any other failure mechanism(s)

NOTE: One alternative to removing representative UT rejected bolts for laboratory examination is to re-inspect all bolts at the next refueling outage if continued operation for one additional fuel cycle can be supported by a technical evaluation. Other possible options such as replacement of a predefined bolt pattern may also be pursued.

- Based on the results of UTS bolt UT inspection and laboratory test results, perform an evaluation to assess future UTS bolt failure potential. The changes to the peak stress at the bolt head-to-shank fillet region as a result of the identified failures should be included for evaluation of increased susceptibility to SCC.

- Incorporate the effect of future UTS bolt failure into the operability evaluation and re-inspection requirement

The general methodology to be used for acceptance criteria for the locking devices will be development of an NDE inspection standard that contains examples of acceptable and unacceptable locking device visual indications. The acceptance of locking devices is evaluated in two ways: a) observations with "failed" bolts and b) observations with all bolts intact. Observations of damaged locking devices with all bolts intact represent a condition very different from that of locking device damage at a bolt location that is failed. In addition, a damaged or missing welded locking clip versus a crimpled locking cup potentially represents different initiating phenomena that need to be evaluated.

Analytical efforts for the UTS bolt failures could be performed on a generic basis for all units except TMI-1, although use of unit-specific loadings could reduce the conservatism for some units. The NDE inspection standard could also be developed generically.

Existing Documentation:

An NRC-accepted crack growth rate for Alloy A-286 or Alloy X-750 material is not currently available. However, the PWROG project (PA-MSC-350) has identified some CGR data that is currently available for a feasibility study of a life assessment approach, if desired.

Surveillance Specimen Holder Tube Studs/Nuts or Bolts and Locking Devices

Item	Applicability	Effect (Mechanism)	Primary Link	Examination Method	Examination Coverage
Core Barrel Assembly Surveillance specimen holder tube (SSHT) studs/nuts (CR-3) or bolts (DB) and their locking devices	CR-3, DB	Cracking (SCC)	UCB bolts LCB bolts	Volumetric examination (UT) of studs or bolts. Visual (VT-3) examination of bolt locking devices or nuts on the 10 year ISI interval.	100% of accessible bolts.

There is a potential for intergranular stress corrosion cracking (IGSCC) of Alloy A-286 and Alloy X-750 bolting. Past B&W failure history exists with the original Alloy A-286 bolt materials in B&W-design units and with applications of Alloy X-750 material within the nuclear industry (in general). Currently, there are no known failures with any of the replacement bolts (Alloy A-286 or Alloy X-750) in the operating B&W-design units or with the original Alloy X-750 (installed at TMI-1 only) in service in the operating B&W-design units.

Component Item Function

The SSHT bolts fasten the surveillance specimen holder tubes to the thermal shield. Failure would result in loosening or dropping of the holder tube to the bottom of the vessel. The SSHT bolts do not have a core support safety function.

Observable Effects:

A volumetric examination (UT) of the bolts and a visual (VT-3) examination of the bolt locking devices

Mockups and qualification efforts exist from the PWROG work (PA-MSC-350) and additional Duke Energy efforts in 2007-2008.

Cracking of the bolts is the main concern and the locking devices are to be examined to identify if any are distorted, loose, broken, or missing.

The PWROG work (PA-MSC-350) also evaluated the potential information that could be determined from only a visual examination of the bolt and locking devices (see AREVA document 51-9081184-001).

Possible Examination Outcomes:

Cracking is anticipated to occur at the head-to-shank area of the bolt where the peak tensile stress exists (i.e., a SCF exists) and OE has shown them to crack at this location in the past, although it may also occur in the shank or stud thread region where high tensile stress is possible too.

- No relevant conditions identified
- Relevant conditions (i.e., crack-like indications, either completely cracked or partially cracked; or non-interpretable UT indications, such as no back wall reflection or multiple reflections with no crack-like indication that is most likely caused by a large or duplex grain size) are identified
  - One or a few studs/nuts or bolts (exact number is unit-specific) are identified with relevant indications
  - More than a few studs/nuts or bolts (exact number is unit-specific) are identified with relevant indications

Locking Devices

- No relevant conditions identified
- One or two are identified with damage or are missing
- More than two bolt locking devices are identified with damage or are missing

Methodology and Data Requirements:

The general analytical methodology to be used for acceptance criteria for the studs/nuts or bolts involves the following steps and inputs if relevant conditions have been identified in the SSHT studs/nuts or bolts:

- A thermal analysis is to be performed
  - Determines bolt temperatures and temperature gradients for normal operating conditions
- A structural analysis is to be performed in which failed studs/nuts or bolts are inactive
  - Stress concentration factors are calculated to determine the peak stresses at the bolt head-to-shank fillet region or stud/nut thread region under normal operating conditions
  - Analysis is performed for all loads and load combinations required for an ASME evaluation (stress limits for threaded structural fasteners in subsection NG and Appendix F)
  - An evaluation of joint stability (or, openness) is also to be performed
- Representative rejected SSHT studs/nuts or bolts are to be removed for laboratory testing to confirm the UT inspection results and IGSCC mechanism, or to identify any other failure mechanism(s)

NOTE: One alternative to removing representative UT rejected bolts for laboratory examination is to re-inspect all bolts at the next refueling outage if continued operation for one additional fuel cycle can be supported by a technical evaluation. Other possible options such as replacement of a predefined bolt pattern may also be pursued.

- Based on the results of SSHT bolt UT inspection and laboratory test results, perform an evaluation to assess future SSHT stud/nut or bolt failure potential. The changes to the peak stress at the bolt head-to-shank fillet region or stud/nut thread region as a result of the identified failures should be included for evaluation of increased susceptibility to SCC.

- Incorporate the effect of future SSHT stud/nut or bolt failure into the operability evaluation and re-inspection requirement

The general methodology to be used for acceptance criteria for the locking devices will be development of an NDE inspection standard that contains examples of acceptable and unacceptable locking device visual indications. The acceptance of locking devices is evaluated in two ways: a) observations with "failed" studs/nuts or bolts and b) observations with all studs/nuts or bolts intact. Observations of damaged locking devices with all studs/nuts or bolts intact represent a condition very different from that of locking device damage at a stud/nut or bolt location that is failed. In addition, a damaged or missing welded locking clip versus a crimpled locking cup potentially represents different initiating phenomena that need to be evaluated.

Since there are only two units, and one has studs/nuts and the other has bolts, unit-specific analyses are required.

Existing Documentation:

An NRC-accepted crack growth rate for Alloy A-286 or Alloy X-750 material is not currently available. However, the PWROG project (PA-MSC-350) has identified some CGR data that is currently available for a feasibility study of a life assessment approach, if desired.

Core Barrel Cylinder

Item	Applicability	Effect (Mechanism)	Primary Link	Examination Method	Examination Coverage
Core Barrel Assembly Core barrel cylinder (including vertical and circumferential seam welds)	All plants	Cracking (IE), including readily detectable cracking	Baffle plates	Justify by evaluation or by replacement.	Inaccessible. See MRP-231 Figure 3-2.

The core barrel cylinders and welds are subject to irradiation embrittlement, which if a flaw would be present and they are subjected to loading that exceeds the materials degraded fracture toughness, such a condition could potentially lead to cracking. There is no known history of OE for cracking of core barrel cylinder and weld material in PWR reactor vessel internals applications.

Component Item Function

Degradation of the core barrel cylinders and welds could result in increased core bypass flow and a reduction in margin to DNB, but would probably have a negligible effect on unit operations and would not be observed except by direct examination. The core barrel supports the fuel assemblies, lower grid, flow distributor, and incore instrument guide tubes. The primary function of the core barrel cylinders and welds during normal power operation is to provide a flow envelope for the core and, thereby limit core bypass flow.

The core barrel cylinders and welds therefore do not have a direct core support safety function; however, they do have a safety function to control bypass around the core during a loss-of-coolant-accident (LOCA).

Observable Effects:

The core barrel cylinders and welds are mostly inaccessible without disassembly. Therefore, no examination is currently required in MRP-227.

The core barrel upper flange-to-core barrel wed and upper HAZ area is partially accessible and could potentially be VT-3 examined.

Nothing is being examined at this time.

Methodology and Data Requirements:

The general analytical methodology to be used for acceptance criteria for the core barrel cylinders and welds involves the following steps and inputs:

- Confirmation of required loading and combination requirements
- Perform a linear-elastic fracture mechanics (LEFM) evaluation to determine the critical crack size using the MRP-211 fracture toughness values
  - A flaw handbook could also be developed
  - Or, justify the existing calculations in MRP-210
- Perform a bypass analysis to justify that sufficient DNB exists in the degraded condition
- An operability evaluation to operate at least one cycle based on possible degradation of the core barrel cylinders and welds should be performed
- An evaluation of the consequences of leaving cracked core barrel cylinders and welds in place during an inspection or replacement campaign should be performed

Analytical efforts could be performed on a generic basis.

Existing Documentation:

- CLB loadings (normal and faulted condition) are available, but a records search may need to be performed to identify them
- No CGRs currently exist
  - CGRs for BWR HWC can be assumed for initial studies

Former Plates

Item	Applicability	Effect (Mechanism)	Primary Link	Examination Method	Examination Coverage
Core Barrel Assembly Former plates	All plants	Cracking (IE), including readily detectable cracking	Baffle plates	Justify by evaluation or by replacement.	Inaccessible. See MRP-231 Figure 3-2.

Former plates are subject to irradiation embrittlement, which if a flaw would be present and they are subjected to loading that exceeds the materials degraded fracture toughness; such a condition could potentially lead to cracking. There is no known history of OE for cracking of former plate material in PWR reactor vessel internals applications.

Component Item Function

The former plates do not have a direct core support safety function; however, they do have a safety function to help maintain the structural integrity of the core barrel assembly during operating conditions.

Observable Effects:

The former plates are mostly inaccessible without disassembly. Therefore, no examination is currently required in MRP-227.

The former plates are partially accessible through openings in the core barrel assembly and could potentially be VT-3 examined.

Nothing is required for examination at this time.

Methodology and Data Requirements:

The general analytical methodology to be used for acceptance criteria for the former plates involves the following steps and inputs:

- Confirmation of required loading and combination requirements
- Perform a linear-elastic fracture mechanics (LEFM) evaluation to determine the critical crack size using the MRP-211 fracture toughness values
  - A flaw handbook could also be developed
  - Or, justify the existing calculations in MRP-210

- An operability evaluation to operate at least one cycle based on possible degradation of the former plates should be performed

- An evaluation of the consequences of leaving cracked former plates in place should be performed

Analytical efforts could be performed on a generic basis.

<u>Existing Documentation:</u>

- CLB loadings (normal and faulted condition) are available, but a records search may need to be performed to identify them

- No CGRs currently exist

  - CGRs for BWR HWC can be assumed for initial studies

Baffle-to-Baffle Bolts

Item	Applicability	Effect (Mechanism)	Primary Link	Examination Method	Examination Coverage
Core Barrel Assembly Baffle-to-baffle bolts	All plants	Cracking (IASCC, IE, IC/ISR/ Fatigue/Wear, Overload)	Baffle-to-former bolts	Internal baffle-to-baffle bolts: No examination requirements, justify by evaluation or by replacement.	Acceptable examination technique not currently available. See MRP-231 Figure 3-2.
				External baffle-to-baffle bolts: No examination requirements, justify by evaluation or by replacement.	Inaccessible. See MRP-231 Figure 3-2.

There is a potential for failure in the form of cracking of baffle-to-baffle bolting to occur mainly from irradiation-assisted stress corrosion cracking (IASCC), but also as a result of irradiation embrittlement, irradiation creep/stress relaxation (leading to fatigue and wear), or overload (from a prying effect). Past failure history exists with baffle-to-former bolt materials (Type 316CW and Type 347) and core barrel-to-former bolt materials (Alloy X-750 and Type 316Ti CW) in non-B&W-design units. Currently, there are no known failures with any of the bolts (Type 304) in the operating B&W-design units. CR-3 has observed what appear to possibly be failed internal baffle-to-baffle bolts, but confirmation has never been made.

<u>Component Item Function</u>

The core barrel assembly consists of the core barrel cylinder, former plates, and baffle plates connected by bolted joints that include: (1) core barrel-to-former bolts (CF bolts), (2) baffle-to-former bolts (BF bolts), and (3) baffle-to-baffle bolts (BB bolts). The core barrel assembly supports the fuel assemblies, lower grid, flow distributor, and incore instrument guide tubes. The baffle plates, former plates, and their joints (including BF bolts) do not have a core support function and are categorized as internals structures. The primary function of the baffle plates, former plates, and their bolted connections is to provide a flow envelope surrounding the core. Also, since they are bolted to the core barrel cylinder, the baffle plates and former plates will produce a small increase on the stiffness and natural frequencies of the core barrel assembly.

The CF and BF bolts have the function of maintaining structural integrity of the baffle and former portion of the structural assembly and thus of maintaining flow geometry during normal operation. For faulted events, a small number of the CF and BF bolts are needed to restrain the baffle so that a coolable core geometry is maintained. The BB bolts are not required for these functions but rather serve to minimize gaps between baffle plates. The BB bolts therefore do not have a core support safety function.

BB bolts are divided into two groups; those BB bolts on internal corners receive neutron fluence that is much higher than those on external corners. The two groups also differ in accessibility for inspection.

Observable Effects:

Cracking of the bolts is the main concern.

No examinations are required at this time in MRP-227.

Methodology and Data Requirements:

The general analytical methodology to be used for acceptance criteria for the BB bolts involves the following steps and inputs:

- A global FEM model is developed to evaluate failures for use in developing the frequency for the I&E guidelines, acceptable failure pattern or numbers, and for use in preparing possible JCOs for the BF and CF bolts
  - Evaluations for these bolt locations will consider BB bolts to be failed and structurally inactive
  - No specific pattern will need to be evaluated as the BB bolts do not perform any core support function, nor are they required to maintain the geometry of the core cavity
  - In addition, no specific acceptance criteria are required for BB bolts
- A hydraulic analysis for evaluation of changes in jet momentum flux due to changes in gaps will be performed to assess changes in jetting and possible fuel failures

Analytical efforts could be performed on all bolt locations on a generic basis.

Existing Documentation:

- A FEM has been developed by the MRP Reactor Internals project to evaluate failures that can be used in evaluating an acceptable failure pattern or number of failed bolts allowed for continued operation, and for use in preparing a JCO

Core Barrel-to-Former Bolts

Item	Applicability	Effect (Mechanism)	Primary Link	Examination Method	Examination Coverage
Core Barrel Assembly Core barrel-to-former bolts	All plants	Cracking (IASCC, IE, IC/ISR/ Fatigue/Wear, Overload)	Baffle-to-former bolts	No examination requirements, justify by evaluation or by replacement.	Inaccessible. See MRP-231 Figure 3-2.

There is a potential for failure in the form of cracking of core barrel-to-former bolting to occur mainly from irradiation-assisted stress corrosion cracking (IASCC), but also as a result of irradiation embrittlement, irradiation creep/stress relaxation (leading to fatigue and wear), or overload (from a prying effect). Past failure history exists with baffle-to-former bolt materials (Type 316CW and Type 347) and core barrel-to-former bolt materials (Alloy X-750 and Type 316Ti CW) in non-B&W-design units. Currently, there are no known failures with any of the bolts (Type 304) in the operating B&W-design units. CR-3 has observed what appear to possibly be failed baffle-to-baffle bolts, but confirmation has never been made.

Component Item Function

The core barrel assembly consists of the core barrel cylinder, former plates, and baffle plates connected by bolted joints that include: (1) core barrel-to-former bolts (CF bolts), (2) baffle-to-former bolts (BF bolts), and (3) baffle-to-baffle bolts (BB bolts). The core barrel assembly supports the fuel assemblies, lower grid, flow distributor, and incore instrument guide tubes. The baffle plates, former plates, and their joints (including BF bolts) do not have a core support function and are categorized as internals structures. The primary function of the baffle plates, former plates, and their bolted connections is to provide a flow envelope surrounding the core. Also, since they are bolted to the core barrel cylinder, the baffle plates and former plates will produce a small increase on the stiffness and natural frequencies of the core barrel assembly.

The CF and BF bolts have the function of maintaining structural integrity of the baffle and former portion of the structural assembly and thus of maintaining flow geometry during normal operation. For faulted events, a small number of the CF and BF bolts are needed to restrain the baffle so that a coolable core geometry is maintained.

Observable Effects:

Cracking of the bolts is the main concern.

No examinations are required at this time in MRP-227.

Methodology and Data Requirements:

The general analytical methodology to be used for acceptance criteria for the CF bolts involves the following steps and inputs:

- A global FEM model is developed to evaluate failures for use in developing the frequency for the I&E guidelines, acceptable failure pattern or numbers, and for use in preparing possible JCOs for the BF or CF bolts and this model can be used to evaluate the need for each of these two bolt locations
- The following inputs are required:
  - Failed or missing CF bolt locations are required for input
  - Thermal input including gamma-heating for design (short-term) and long-term operating conditions
  - Irradiated material property input as a function of aging (EFPY) of the core barrel assembly
  - Applicable test data to establish stress and/or strain and fatigue strength limits of the CF bolts in addition to the licensing basis requirements
  - Faulted load licensing basis requirements based on existing evaluations and modified as needed
  - Acceptable baffle displacements, changes in flow slot or gaps, and/or changes in natural frequency of the baffle/former structure, as applicable to the BF and BB bolts
- Appropriate structural evaluations are performed to demonstrate the above acceptance criteria
- If necessary, the existing model will be modified to be suitable for dynamic loadings such as imposed core barrel motion due to turbulence induced vibration

Analytical efforts could be performed on all bolt locations on a generic basis.

Existing Documentation:

- A FEM has been developed by the MRP Reactor Internals project to evaluate failures that can be used in evaluating an acceptable failure pattern or number of failed bolts allowed for continued operation, and for use in preparing a JCO
  - The FEM model for the MRP project has not been applied to dynamic loadings
- Past efforts however have assumed all CF bolts to be intact
- No acceptable evaluation or analysis has been completed to date for determining a re-inspection schedule

Locking Devices for External Baffle-to-Baffle and Core Barrel-to-Former Bolts

Item	Applicability	Effect (Mechanism)	Primary Link	Examination Method	Examination Coverage
Core Barrel Assembly Locking devices, including locking welds, for the external baffle-to-baffle bolts and core barrel-to-former bolts	All plants	Cracking (IASCC, IE)	Locking devices, including locking welds, of baffle-to-former bolts or internal baffle-to-baffle bolts	Justify by evaluation or by replacement.	Inaccessible. See MRP-231 Figure 3-2.

There is a potential for irradiation-assisted stress corrosion cracking (IASCC) of the locking devices or welds for the external baffle-to-baffle and core barrel-to-former bolting. There is also the potential for failure by overload for these locking devices and welds due to slip between the bolts and plates (see MRP-231). Past failure history exists with cracked and missing internal baffle-to-baffle bolt locking devices and cracked locking device welds in B&W-design units.

Component Item Function

The locking devices and welds are not normally loaded unless the bolt is broken or loose. Loading of the locking devices and welds could also occur due to the slip between the bolt and plate. The locking devices and welds have no core support safety function.

Observable Effects:

Items are inaccessible and no known technique is available other than disassembly of the core barrel assembly for a visual examination; therefore, no examinations are required at this time in MRP-227.

Cracking of the locking devices or welds is the concern.

Methodology and Data Requirements:

Locking device failure in itself is not a safety concern and an assessment can be prepared stating this as such. Failure of the bolting locations is of more concern and is covered in the bolting summary pages.

Analytical efforts could be performed on a generic basis for the B&W units.

Existing Documentation:

Nothing is available at this time.

Lower Grid Fuel Assembly Support Pad Items

Item	Applicability	Effect (Mechanism)	Primary Link	Examination Method	Examination Coverage
Lower Grid Assembly Lower grid fuel assembly support pad items: pad, pad-to-rib section welds, Alloy X-750 dowel, cap screw, and their locking welds (Note: The pads, dowels, and cap screws are included because of TE/IE of the welds)	All plants	Cracking (IE), including the detection of separated or missing welds, missing support pads, dowels, cap screws and locking welds, or misalignment of the support pads.	IMI guide tube spiders Spider-to-lower grid rib section welds	Visual (VT-3) examination.	100% of accessible pads, dowels, and cap screws, and associated welds. See MRP-231 Figure 3-6.

Lower grid fuel assembly support pad items are mostly subject to irradiation embrittlement, with some also susceptible to thermal aging and irradiation embrittlement, which if a flaw would be present and they are subjected to loading that exceeds the materials degraded fracture toughness, such a condition could potentially lead to cracking. There is no known history of OE for cracking of these materials in PWR reactor vessel internals applications.

Component Item Function

These welds serve as loose part prevention devices and are not structural. Small cracks in the locking weld are acceptable since the locking function can be maintained as long as any part of the weld is present. The fuel assembly support pads serve as guidance for loading of the fuel into the core. Once the fuel assemblies are loaded into the core, the support pads no longer have any function.

Therefore, the lower grid fuel assembly support pad items have no core support safety function.

Observable Effects

A visual (VT-3) examination of the lower grid fuel assembly support pad items is to be performed.

Loss of the lower grid fuel assembly support pad is the main concern and therefore the various items are to be examined to identify if any are separated or missing, if a dowel is missing, or the support pad is misaligned (clearly out of perpendicularity).

Possible Examination Outcomes:

- No relevant conditions exist
- A single weld is separated or missing
- Several welds are separated or missing
- A single dowel is missing
- Several dowels are missing
- A support pad is misaligned (clearly out of perpendicularity) or missing
- Several support pads are misaligned (clearly out of perpendicularity) or missing

Methodology and Data Requirements:

The general methodology to be used for acceptance criteria for the lower grid fuel assembly support items will be development of an NDE inspection standard that contains examples of acceptable and unacceptable lower grid fuel assembly support pad item visual indications. The function of the support pad can be maintained as long as a portion of any of the welds is in place. Significant cracking of the welds and subsequent loss of the dowel does not compromise the function of the fuel assembly support pad unless the screw also fails.

The following items will be examined to establish VT-3 acceptance criteria and the technical justification:

- Identify normal and faulted operating loads for the fuel assembly support pad dowels

- Evaluate the consequences of leaving partially cracked locking welds securely in place during an inspection

- Identify the areas to be examined containing what is rejectable and requiring further evaluation

- Develop repair strategies for leaving in place if secured from being a loose part or removal and replacement activities

A UT examination of the fuel assembly support pad screw or a technical justification for removal of the fuel assembly support pad may be necessary.

Analytical efforts could be performed on a generic basis for the B&W units. The NDE inspection standard could also be developed generically.

Existing Documentation:

Minimal information is currently available.

Lower Grid Shock Pad Bolts and Locking Devices

Item	Applicability	Effect (Mechanism)	Primary Link	Examination Method	Examination Coverage
Lower Grid Assembly Lower grid shock pad bolts and their locking devices	TMI-1	Cracking (SCC)	UCB bolts LCB bolts	Volumetric examination (UT). Visual (VT-3) examination of bolt locking devices on the 10-year ISI interval.	100% of accessible bolts. See MRP-231 Figure 3-4

There is a potential for intergranular stress corrosion cracking (IGSCC) of Alloy A-286 and Alloy X-750 bolting. Past B&W failure history exists with the original Alloy A-286 bolt materials in B&W-design units and with applications of Alloy X-750 material within the nuclear industry (in general). Currently, there are no known failures with any of the replacement bolts (Alloy A-286 or Alloy X-750) in the operating B&W-design units or with the original Alloy X-750 (installed at TMI-1 only) in service in the operating B&W-design units.

Component Item Function

The function of the lower grid shock pad bolts is to fasten the shock pads to the lower grid assembly. Shock pads must be in place to carry accidental core drop loads. The bolts do not function to carry the core drop load, but serve to hold the pad in place. Each shock pad is held by two bolts. At least one must be intact on each shock pad to prevent a loose part.

The shock pad bolts are also part of the joint between the lower end of the thermal shield cylinder and the lower grid assembly. At TMI-1, these bolts are fabricated from Alloy X-750 material and are designed to hold the shock pad in place and engage the lower thermal shield too. Hence, the shock pad bolts at TMI-1 also function as LTS bolts. This is a unique design feature not shared by the other B&W units. The lower thermal shield joint acts as a restraint to vertical and rotational motion of the bottom of the thermal shield, but does not act as a direct core support component and therefore does not have a core support safety function. Evaluation of the lower thermal shield joint is based on 96 LTS bolts with no credit taken for strength of the shock pad bolts. Therefore, the only function to be maintained by the shock pad bolts is to keep the shock pads in place.

Observable Effects:

A volumetric examination (UT) of the bolts and a visual (VT-3) examination of the bolt locking devices

Mockups and qualification efforts exist from the PWROG work (PA-MSC-350) and additional Duke Energy efforts in 2007-2008.

Cracking of the bolts is the main concern and the locking devices are to be examined to identify if any are distorted, loose, broken, or missing.

The PWROG work (PA-MSC-350) also evaluated the potential information that could be determined from only a visual examination of the bolt and locking devices (see AREVA document 51-9081184-001).

Cracking is anticipated to occur at the head-to-shank area where the peak tensile stress exists (i.e., a SCF exists) and OE has shown them to crack at this location in the past, although it may also occur in the shank thread region where high tensile stress is possible too.

- No relevant conditions identified

- Relevant conditions (i.e., crack-like indications, either completely cracked or partially cracked; or non-interpretable UT indications, such as no back wall reflection or multiple reflections with no crack-like indication that is most likely caused by a large or duplex grain size) are identified

  - If one bolt is missing on a shock pad, it is regarded as relevant since there is no redundancy to hold the pad in place.

  - If two bolts on any individual shock pad are identified with relevant conditions, the shock pad could become a loose part and may not be in place in the event of a core drop accident

<u>Locking Devices</u>

- No relevant conditions identified
- One or two are identified with damage or are missing
- More than two bolt locking devices are identified with damage or are missing

<u>Methodology and Data Requirements:</u>

The general analytical methodology to be used for acceptance criteria for the bolts involves the following steps and inputs if relevant conditions have been identified in the lower grid shock pad bolts:

- If two bolts on any individual shock pad are identified with relevant indications, it is an indication that the shock pad may become loose and will not be in place to carry a core drop. A structural evaluation is to be performed to determine if remaining pads can carry the core drop load or if the load can be carried without the shock pad in place

- If one of two bolts has an indication, an analysis is to be performed to assess loads on the remaining bolt and its potential for future failure

  - Loads on this bolt will include those evaluated as part of modeling of the lower thermal shield joint as described for LTS bolts

  - Stress concentration factors are calculated to determine the peak stresses at the bolt head-to-shank fillet region under normal operating conditions

  - Structural evaluation will be performed to determine peak stress in remaining shock pad bolts for use in assessing potential for future bolt failure

- Representative rejected lower grid shock pad bolts are to be removed for laboratory testing to confirm the UT inspection results and IGSCC mechanism, or to identify any other failure mechanism(s)

NOTE: One alternative to removing representative UT rejected bolts for laboratory examination is to re-inspect all bolts at the next refueling outage if continued operation for one additional fuel cycle can be supported by a technical evaluation. Other possible options such as replacement of a predefined bolt pattern may also be pursued.

- Based on the results of lower grid shock pad bolt UT inspection and laboratory test results, perform an evaluation to assess future lower grid shock pad bolt failure potential. The changes to the peak stress at the bolt head-to-shank fillet region as a result of the identified failures should be included for evaluation of increased susceptibility to SCC.

- Incorporate the effect of future lower grid shock pad bolt failure into the operability evaluation and re-inspection requirement

The general methodology to be used for acceptance criteria for the locking devices will be development of an NDE inspection standard that contains examples of acceptable and unacceptable locking device visual indications. The acceptance of locking devices is evaluated in two ways: a) observations with "failed" bolts and b) observations with all bolts intact. Observations of damaged locking devices with all bolts intact represent a condition very different from that of locking device damage at a bolt location that is failed. In addition, a damaged or missing welded locking clip versus a crimpled locking cup potentially represents different initiating phenomena that need to be evaluated.

A TMI unit-specific analytical effort is required.

<u>Existing Documentation:</u>

An NRC-accepted crack growth rate for Alloy A-286 or Alloy X-750 material is not currently available. However, the PWROG project (PA-MSC-350) has identified some CGR data that is currently available for a feasibility study of a life assessment approach, if desired.

Lower Thermal Shield Bolts and Locking Devices

Item	Applicability	Effect (Mechanism)	Primary Link	Examination Method	Examination Coverage
Lower Grid Assembly Lower thermal shield studs/nuts or bolts (LTS) and their locking devices	All plants	Cracking (SCC)	UCB bolts LCB bolts	Volumetric examination (UT) of studs or bolts. Visual (VT-3) examination of bolt locking devices or nuts on the 10-year ISI interval.	100% of accessible bolts. See MRP-231 Figure 3-8.

There is a potential for intergranular stress corrosion cracking (IGSCC) of Alloy A-286 and Alloy X-750 bolting. Past B&W failure history exists with the original Alloy A-286 bolt materials in B&W-design units and with applications of Alloy X-750 material within the nuclear industry (in general). Currently, there are no known failures with any of the replacement bolts (Alloy A-286 or Alloy X-750) in the operating B&W-design units or with the original Alloy X-750 (installed at TMI-1 only) in service in the operating B&W-design units.

Component Item Function

The LTS bolts fasten the thermal shield cylinder to the lower grid assembly. The LTS joint acts as a restraint to vertical and rotational motion of the bottom of the thermal shield, but does not act as a direct core support component and therefore does not have a core support safety function.

Observable Effects:

A volumetric examination (UT) of the studs/nuts or bolts and a visual (VT-3) examination of the stud/nut or bolt locking devices

Mockups and qualification efforts exist from the PWROG work (PA-MSC-350) and additional Duke Energy efforts in 2007-2008.

Cracking of the bolts is the main concern and the locking devices are to be examined to identify if any are distorted, loose, broken, or missing.

The PWROG work (PA-MSC-350) also evaluated the potential information that could be determined from only a visual examination of the bolt and locking devices (see AREVA document 51-9081184-001).

Possible Examination Outcomes:

Cracking is anticipated to occur at the head-to-shank area of the bolt where the peak tensile stress exists (i.e., a SCF exists) and OE has shown them to crack at this location in the past, although it may also occur in the shank or stud thread region where high tensile stress is possible too.

- No relevant conditions identified
- Relevant conditions (i.e., crack-like indications, either completely cracked or partially cracked; or non-interpretable UT indications, such as no back wall reflection or multiple reflections with no crack-like indication that is most likely caused by a large or duplex grain size) are identified
  - One or a few studs/nuts or bolts (exact number is unit-specific) are identified with relevant indications
  - More than a few studs/nuts or bolts (exact number is unit-specific) are identified with relevant indications

Locking Devices

- No relevant conditions identified
- One or two are identified with damage or are missing
- More than two bolt locking devices are identified with damage or are missing

Methodology and Data Requirements:

The general analytical methodology to be used for acceptance criteria for the studs/nuts or bolts involves the following steps and inputs if relevant conditions have been identified in the LTS studs/nuts or bolts:

- A finite element model (FEM) is to be developed for the local geometry with contact conditions, pretension elements, loads and boundary conditions
- A thermal analysis is to be performed
  - Determines bolt temperatures and temperature gradients for normal operating conditions
- A structural analysis is to be performed in which failed studs/nuts or bolts are inactive
  - Stress concentration factors are calculated to determine the peak stresses at the bolt head-to-shank fillet region or stud/nut thread region under normal operating conditions
  - Analysis is performed for all loads and load combinations required for an ASME evaluation (stress limits for threaded structural fasteners in subsection NG and Appendix F)
  - An evaluation of joint stability (or, openness) is also to be performed

- Representative rejected LTS studs/nuts or bolts are to be removed for laboratory testing to confirm the UT inspection results and IGSCC mechanism, or to identify any other failure mechanism(s)

NOTE: One alternative to removing representative UT rejected bolts for laboratory examination is to re-inspect all bolts at the next refueling outage if continued operation for one additional fuel cycle can be supported by a technical evaluation. Other possible options such as replacement of a predefined bolt pattern may also be pursued.

- Based on the results of LTS bolt UT inspection and laboratory test results, perform an evaluation to assess future LTS stud/nut or bolt failure potential. The changes to the peak stress at the bolt head-to-shank fillet region or stud/nut thread region as a result of the identified failures should be included for evaluation of increased susceptibility to SCC.

- Incorporate the effect of future LTS stud/nut or bolt failure into the operability evaluation and re-inspection requirement

The general methodology to be used for acceptance criteria for the locking devices will be development of an NDE inspection standard that contains examples of acceptable and unacceptable locking device visual indications. The acceptance of locking devices is evaluated in two ways: a) observations with "failed" studs/nuts or bolts and b) observations with all studs/nuts or bolts intact. Observations of damaged locking devices with all studs/nuts or bolts intact represent a condition very different from that of locking device damage at a stud/nut or bolt location that is failed. In addition, a damaged or missing welded locking clip versus a crimpled locking cup potentially represents different initiating phenomena that need to be evaluated.

Due to the variations in stud/nut or bolt materials used and loadings among the units, unit-specific analyses are required. The NDE inspection standard could be developed on a generic basis.

<u>Existing Documentation:</u>

An NRC-accepted crack growth rate for Alloy A-286 or Alloy X-750 material is not currently available. However, the PWROG project (PA-MSC-350) has identified some CGR data that is currently available for a feasibility study of a life assessment approach, if desired.

Flow Distributor Bolts and Locking Devices

Item	Applicability	Effect (Mechanism)	Primary Link	Examination Method	Examination Coverage
Flow Distributor Assembly Flow distributor bolts (FD) and their locking devices	All plants	Cracking (SCC)	UCB bolts LCB bolts	Volumetric examination (UT). Visual (VT-3) examination of bolt locking devices on the 10-year ISI interval.	100% of accessible bolts. See MRP-231 Figure 3-8.

There is a potential for intergranular stress corrosion cracking (IGSCC) of Alloy A-286 and Alloy X-750 bolting. Past B&W failure history exists with the original Alloy A-286 bolt materials in B&W-design units and with applications of Alloy X-750 material within the nuclear industry (in general). Currently, there are no known failures with any of the replacement bolts (Alloy A-286 or Alloy X-750) in the operating B&W-design units or with the original Alloy X-750 (installed at TMI-1 only) in service in the operating B&W-design units.

Component Item Function

The FD bolts fasten the flange on the flow distributor to the lower grid assembly. The joint also clamps the lower grid support plate in place between the bottom of the lower grid assembly and a ledge on the ID of the flow distributor. A clamp ring spans the gap between the bottom mating face of the lower grid assembly and the top of the support plate, providing the compressive force holding the support plate in place. The FD bolts do not have a core support safety function.

Observable Effects:

A volumetric examination (UT) of the bolts and a visual (VT-3) examination of the bolt locking devices

Mockups and qualification efforts exist from the PWROG work (PA-MSC-350) and additional Duke Energy efforts in 2007-2008.

Cracking of the bolts is the main concern and the locking devices are to be examined to identify if any are distorted, loose, broken, or missing.

The PWROG work (PA-MSC-350) also evaluated the potential information that could be determined from only a visual examination of the bolt and locking devices (see AREVA document 51-9081184-001).

Possible Examination Outcomes:

Bolts

Cracking is anticipated to occur at the head-to-shank area where the peak tensile stress exists (i.e., a SCF exists) and OE has shown them to crack at this location in the past, although it may also occur in the shank thread region where high tensile stress is possible too.

- No relevant conditions identified
- Relevant conditions (i.e., crack-like indications, either completely cracked or partially cracked; or non-interpretable UT indications, such as no back wall reflection or multiple reflections with no crack-like indication that is most likely caused by a large or duplex grain size) are identified
  - One or a few bolts (exact number is unit-specific) are identified with relevant indications
  - More than a few bolts (exact number is unit-specific) are identified with relevant indications

Locking Devices

- No relevant conditions identified
- One or two are identified with damage or are missing
- More than two bolt locking devices are identified with damage or are missing

Methodology and Data Requirements:

The general analytical methodology to be used for acceptance criteria for the bolts involves the following steps and inputs if relevant conditions have been identified in the FD bolts:

- A finite element model (FEM) is to be developed for the local geometry with contact conditions, pretension elements, loads and boundary conditions
- A thermal analysis is to be performed
  - Determines bolt temperatures and temperature gradients for normal operating conditions
- A structural analysis is to be performed in which failed bolts are inactive
  - Stress concentration factors are calculated to determine the peak stresses at the bolt head-to-shank fillet region under normal operating conditions
  - Analysis is performed for all loads and load combinations required for an ASME evaluation (stress limits for threaded structural fasteners in subsection NG and Appendix F)
  - An evaluation of joint stability (or, openness) is also to be performed

- Representative rejected FD bolts are to be removed for laboratory testing to confirm the UT inspection results and IGSCC mechanism, or to identify any other failure mechanism(s)

NOTE: One alternative to removing representative UT rejected bolts for laboratory examination is to re-inspect all bolts at the next refueling outage if continued operation for one additional fuel cycle can be supported by a technical evaluation. Other possible options such as replacement of a predefined bolt pattern may also be pursued.

- Based on the results of FD bolt UT inspection and laboratory test results, perform an evaluation to assess future FD bolt failure potential. The changes to the peak stress at the bolt head-to-shank fillet region as a result of the identified failures should be included for evaluation of increased susceptibility to SCC.

- Incorporate the effect of future FD bolt failure into the operability evaluation and re-inspection requirement

The general methodology to be used for acceptance criteria for the locking devices will be development of an NDE inspection standard that contains examples of acceptable and unacceptable locking device visual indications. The acceptance of locking devices is evaluated in two ways: a) observations with "failed" bolts and b) observations with all bolts intact. Observations of damaged locking devices with all bolts intact represent a condition very different from that of locking device damage at a bolt location that is failed. In addition, a damaged or missing welded locking clip versus a crimpled locking cup potentially represents different initiating phenomena that need to be evaluated.

Analytical efforts for the FD bolts could be performed on a generic basis (for all units except TMI-1), although unit-specific analyses could decrease the conservatism for some units. The NDE inspection standard could also be developed generically.

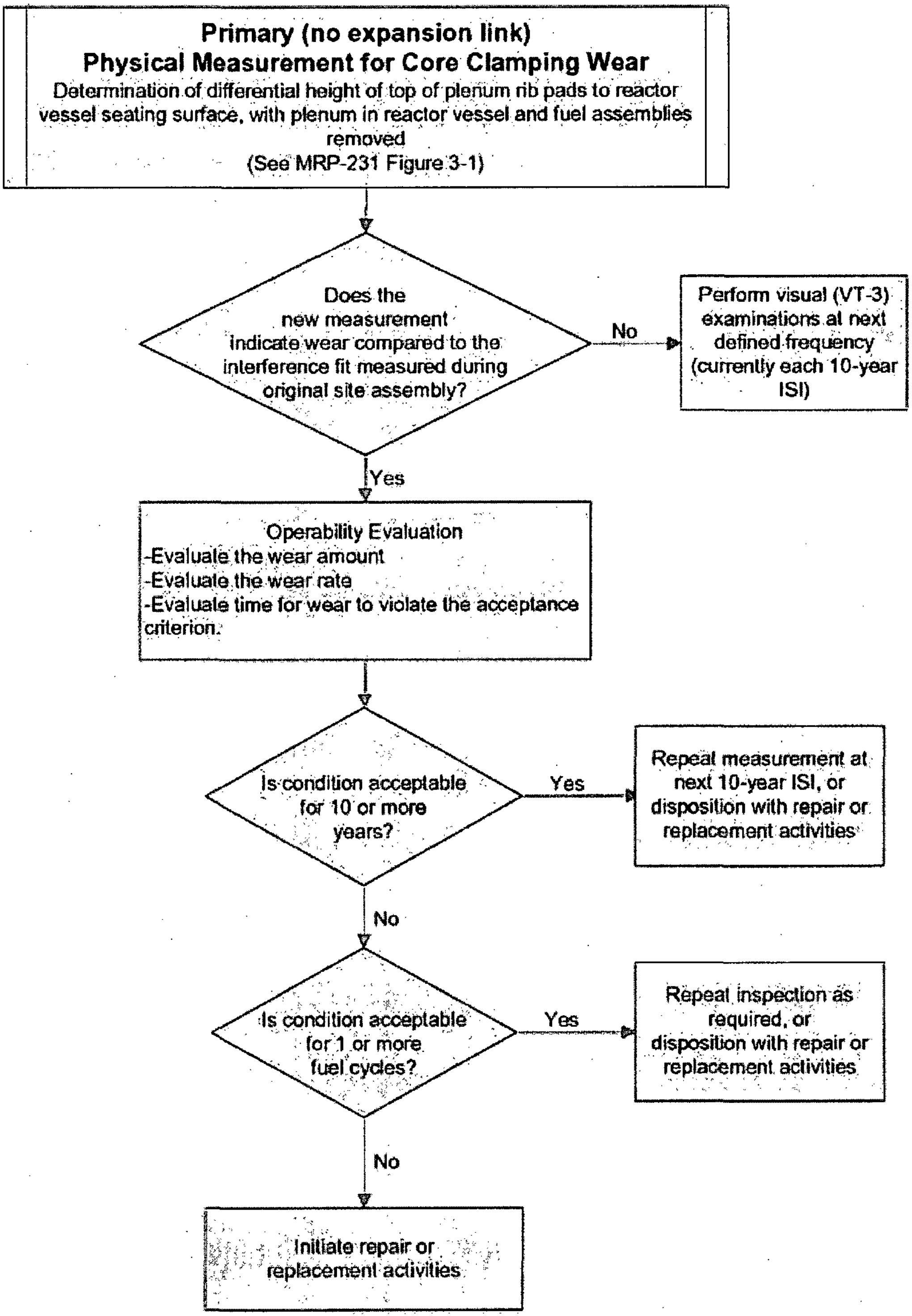
<u>Existing Documentation:</u>

A generic flow distributor bolt stress analysis (for all units except TMI-1) was developed for the MRP reactor internals project in 2007 (see AREVA NP document 32-9059506).

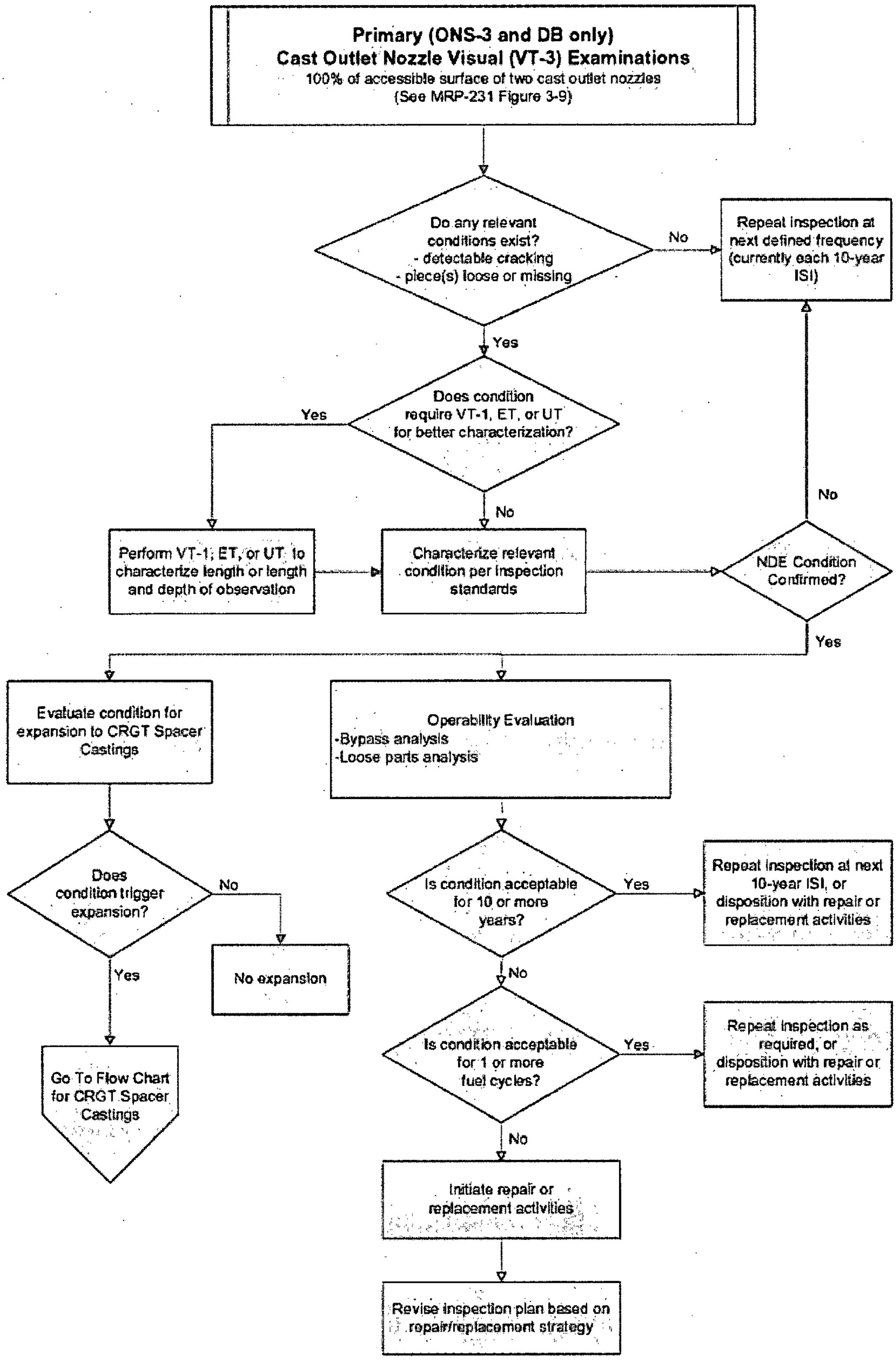
An NRC-accepted crack growth rate for Alloy A-286 or Alloy X-750 material is not currently available. However, the PWROG project (PA-MSC-350) has identified some CGR data that is currently available for a feasibility study of a life assessment approach, if desired.

APPENDIX B
ACCEPTANCE CRITERIA FLOWCHARTS FOR B&W-DESIGNED COMPONENTS INCLUDED IN MRP-227

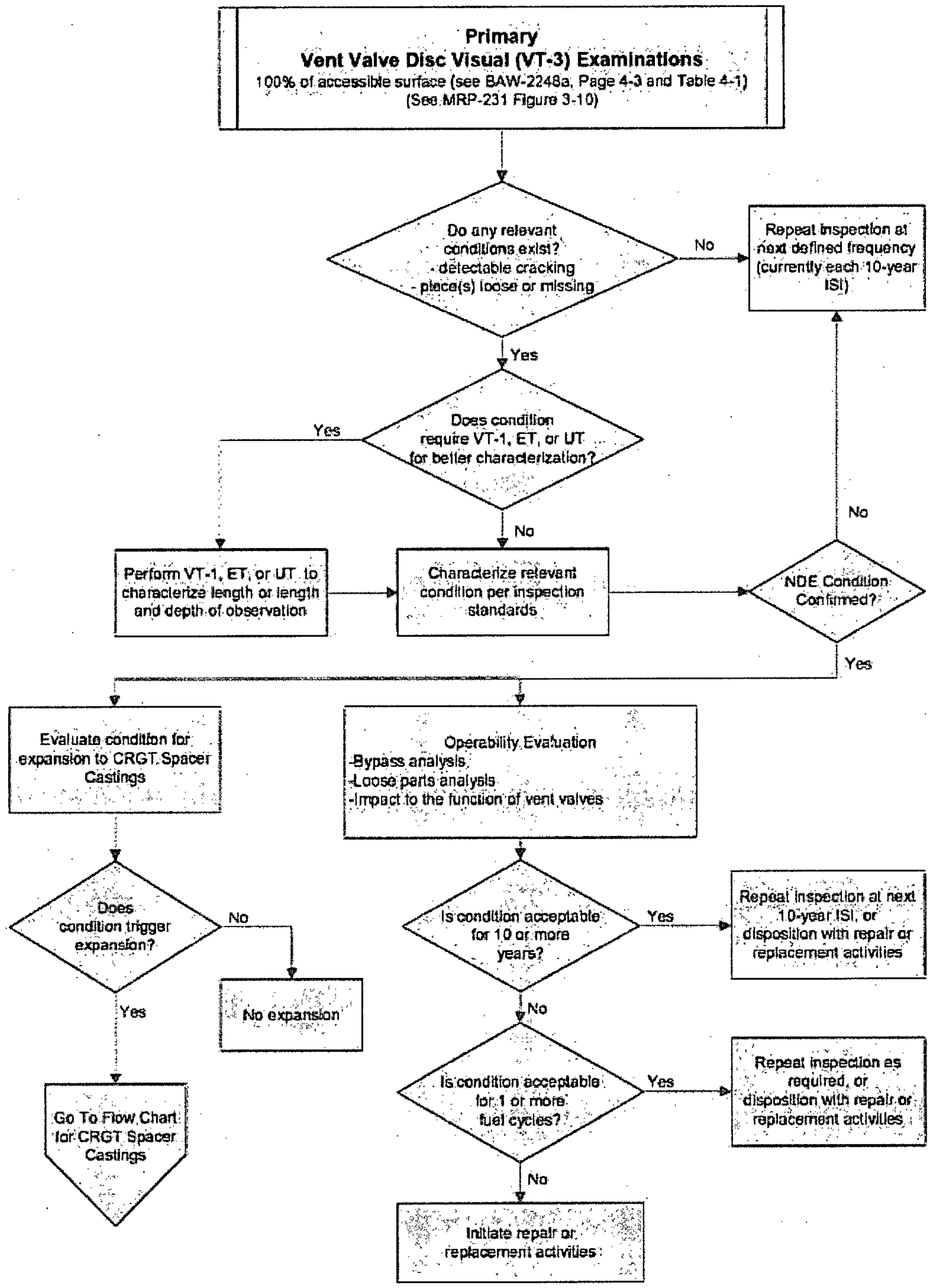
B.1.1 Core Clamping Items



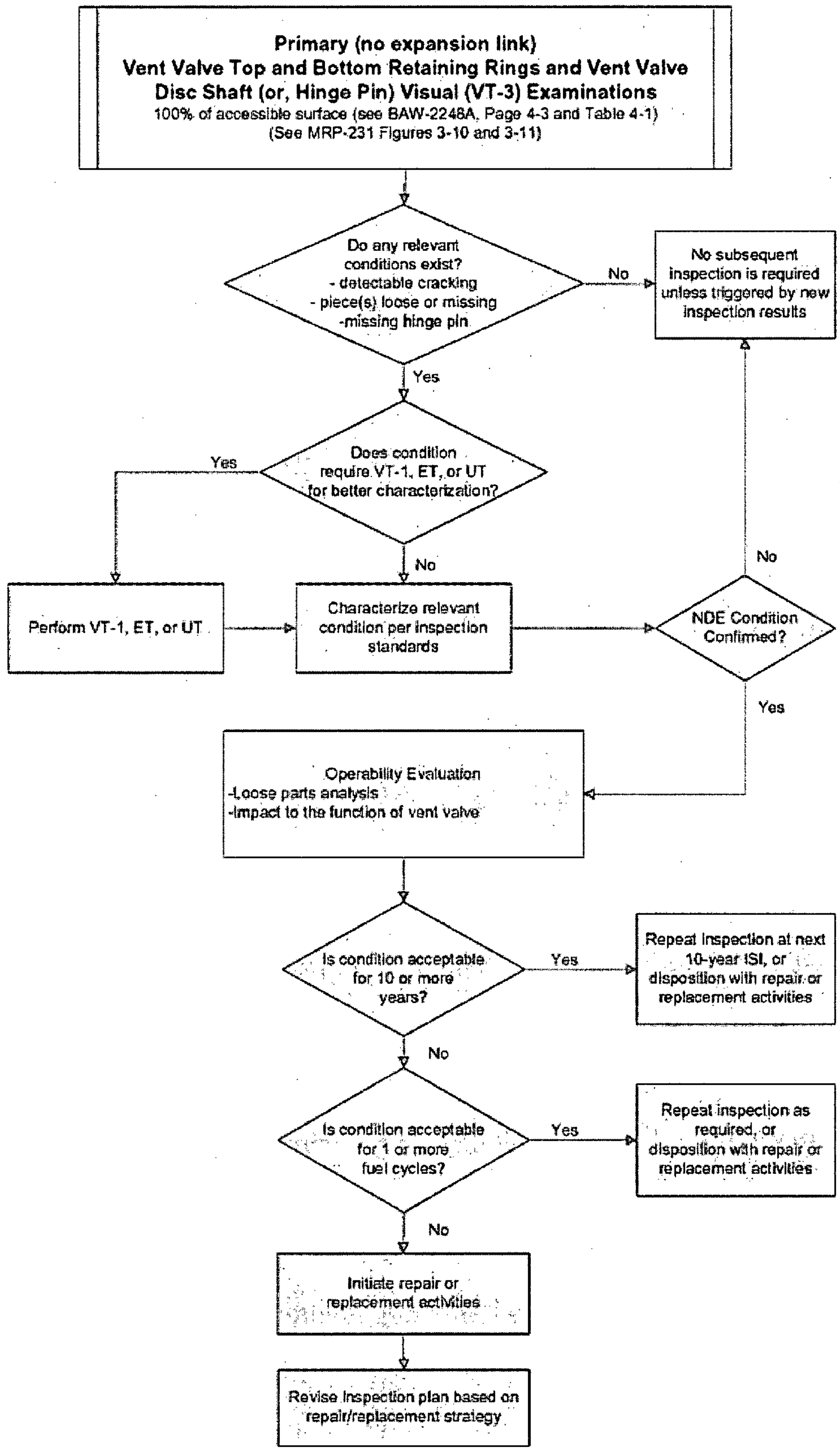
B.1.2 Core Support Shield Cast Outlet Nozzles



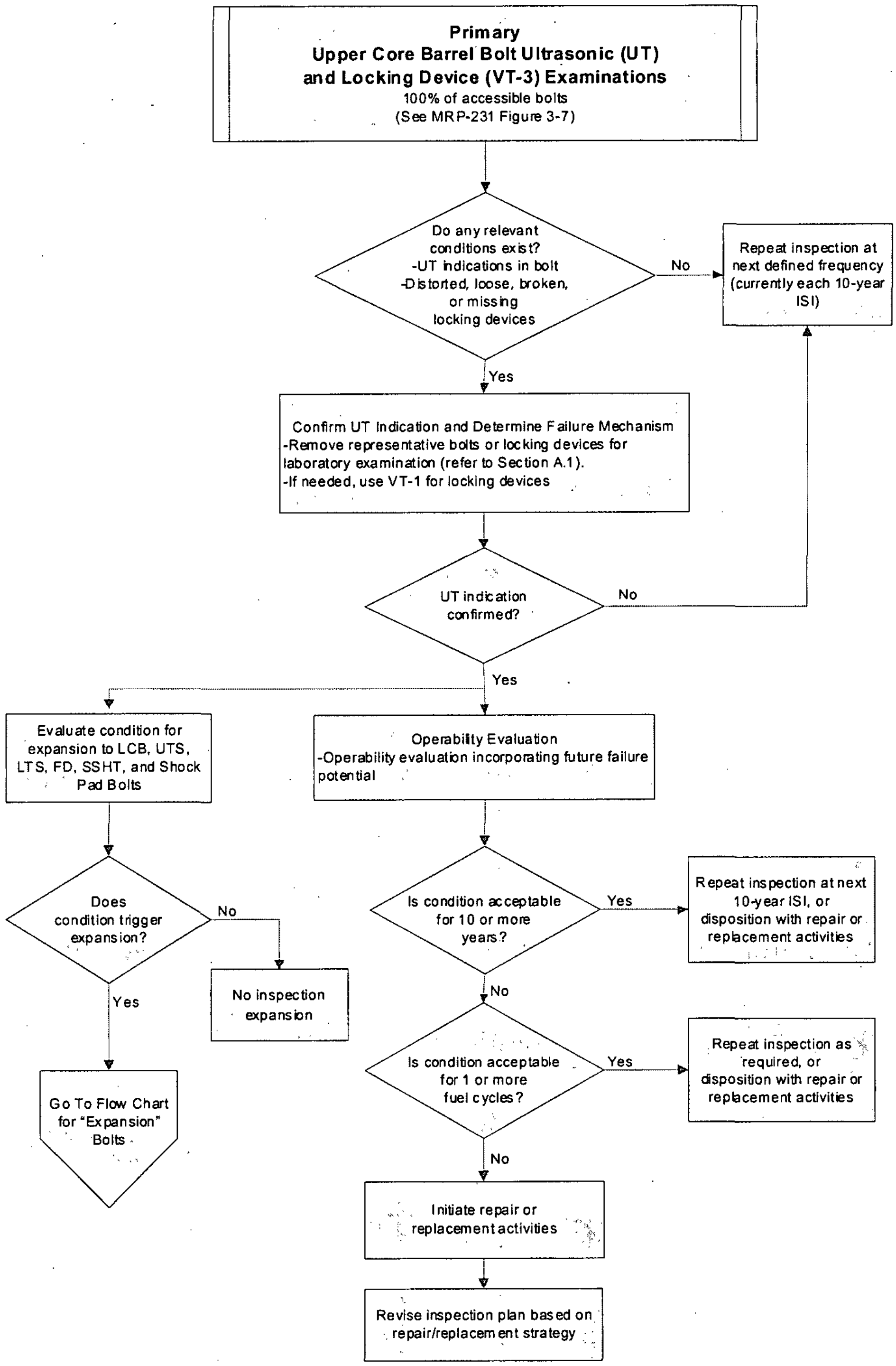
B.1.3 Core Support Shield Vent Valve Discs



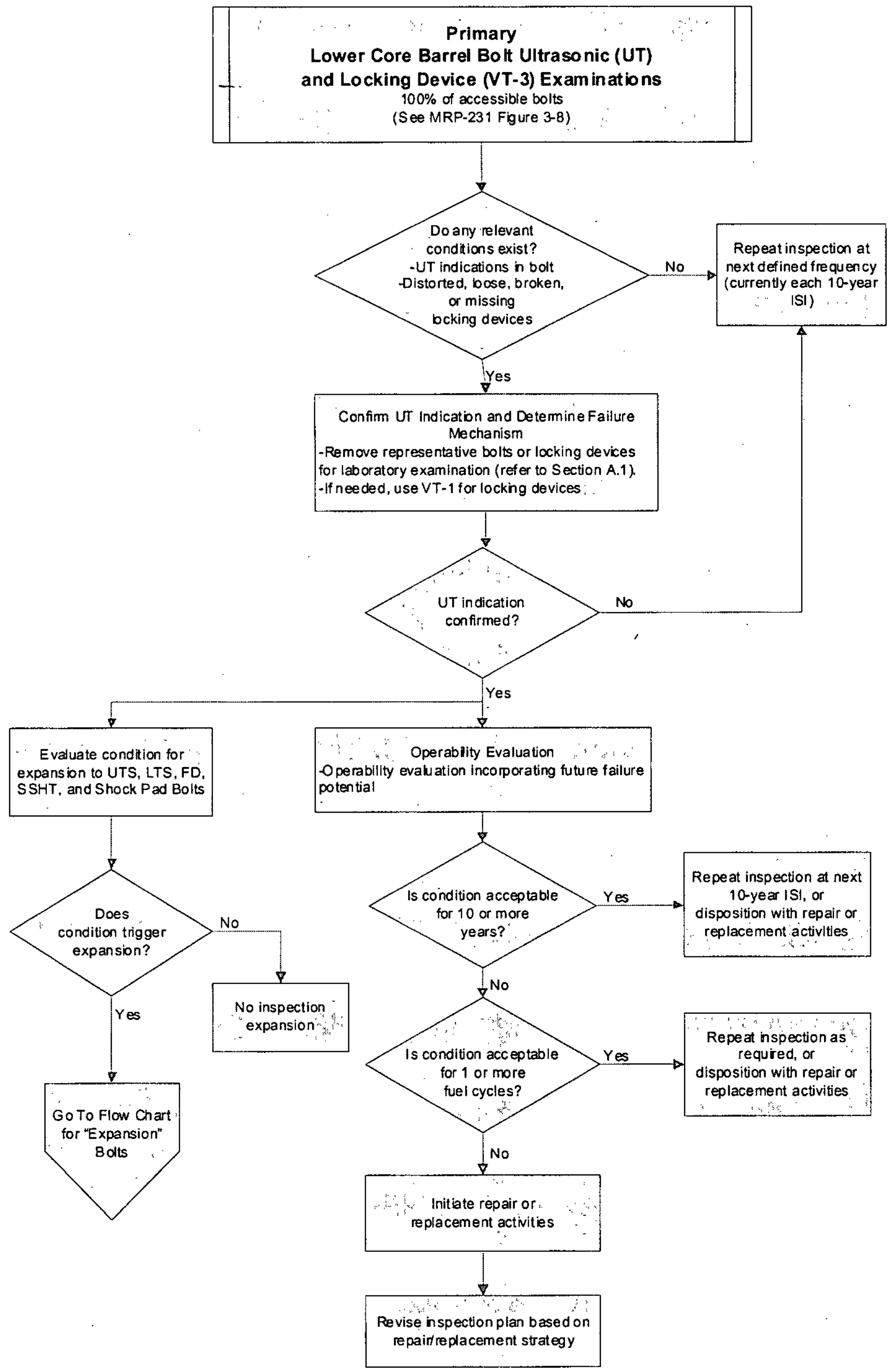
B.1.4 Core Support Shield Vent Valve Retaining Rings and Disc Shaft



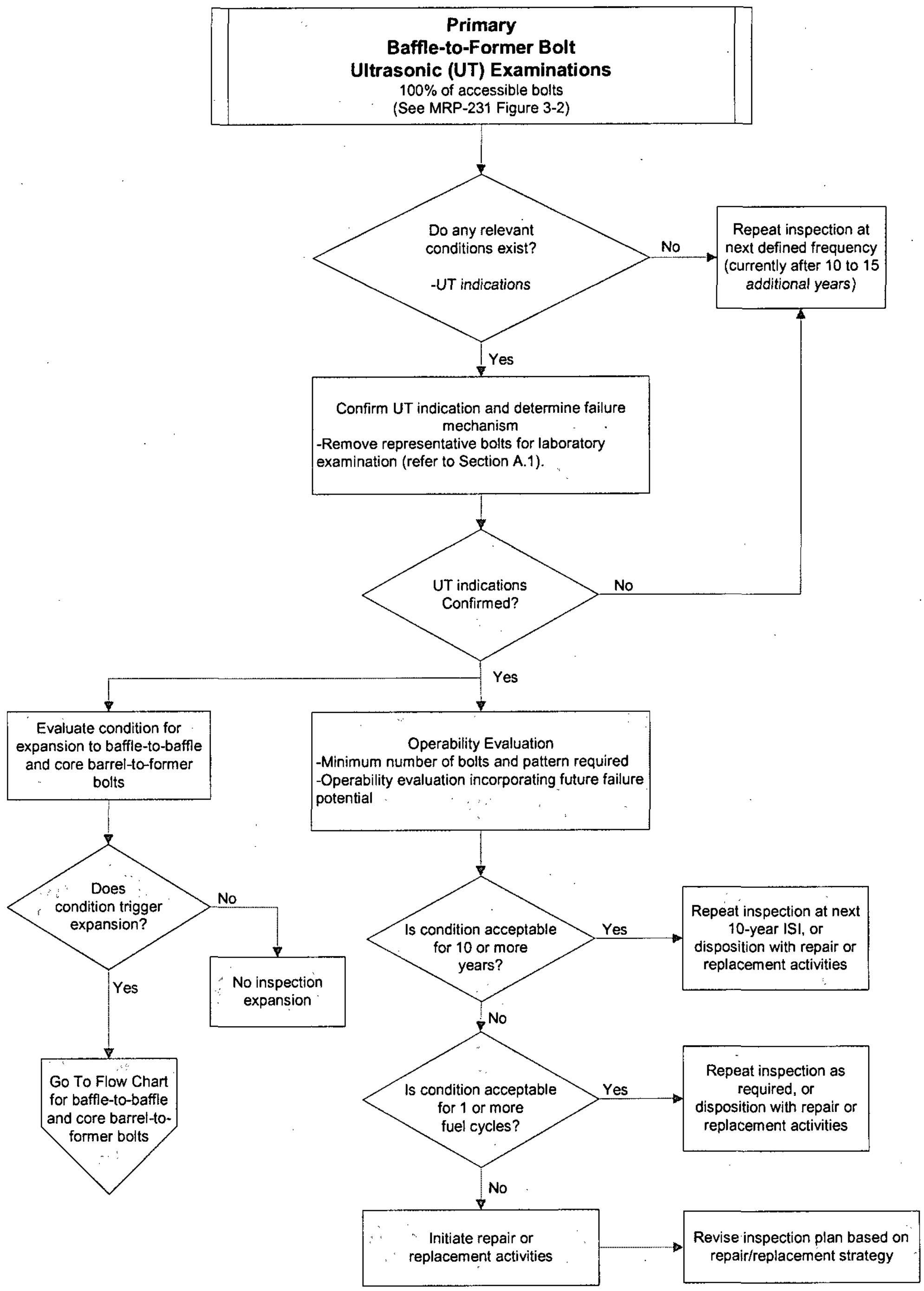
B.1.5 Upper Core Barrel Bolts and Locking Devices



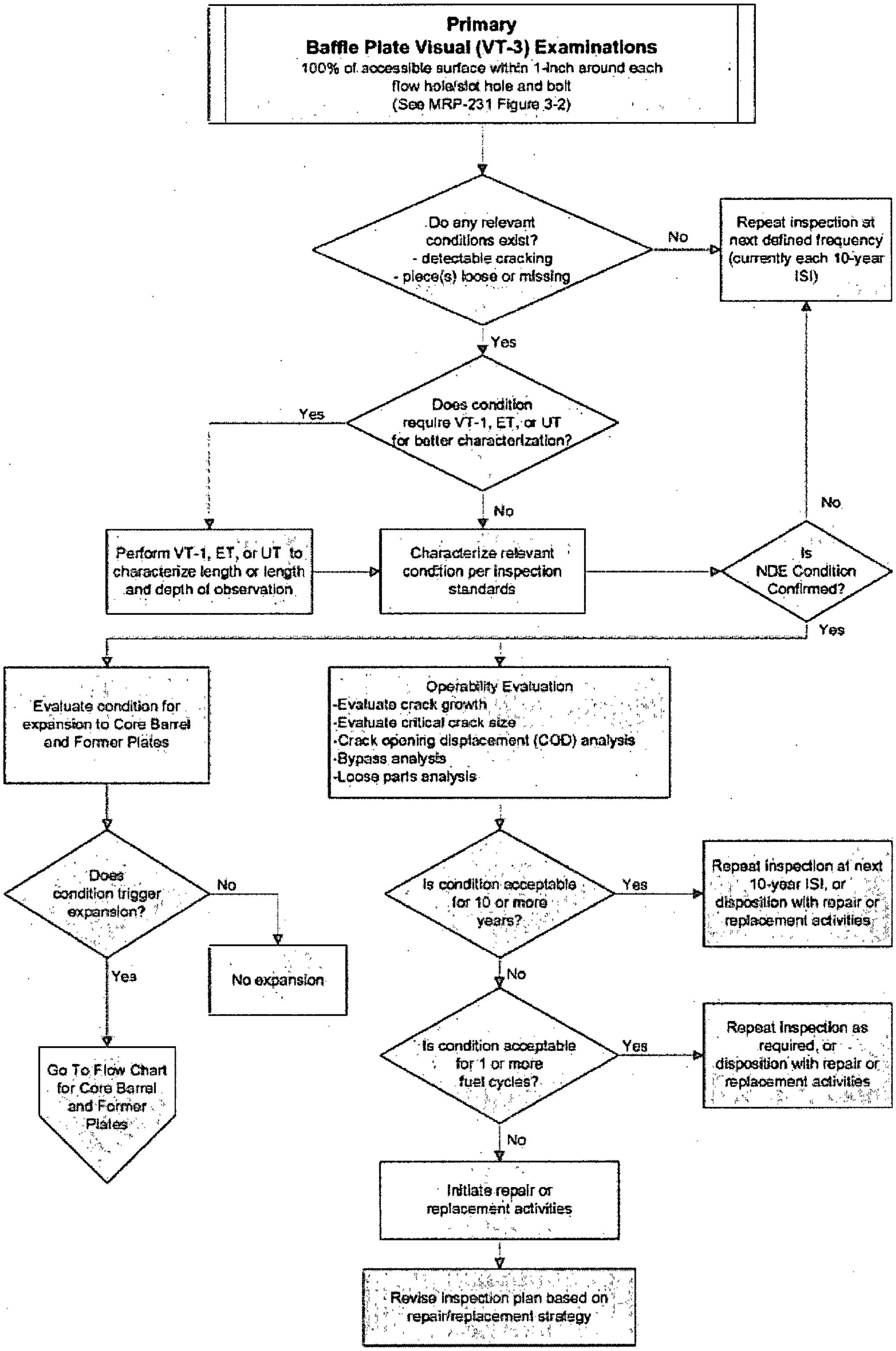
B.1.6 Lower Core Barrel Bolts and Locking Devices



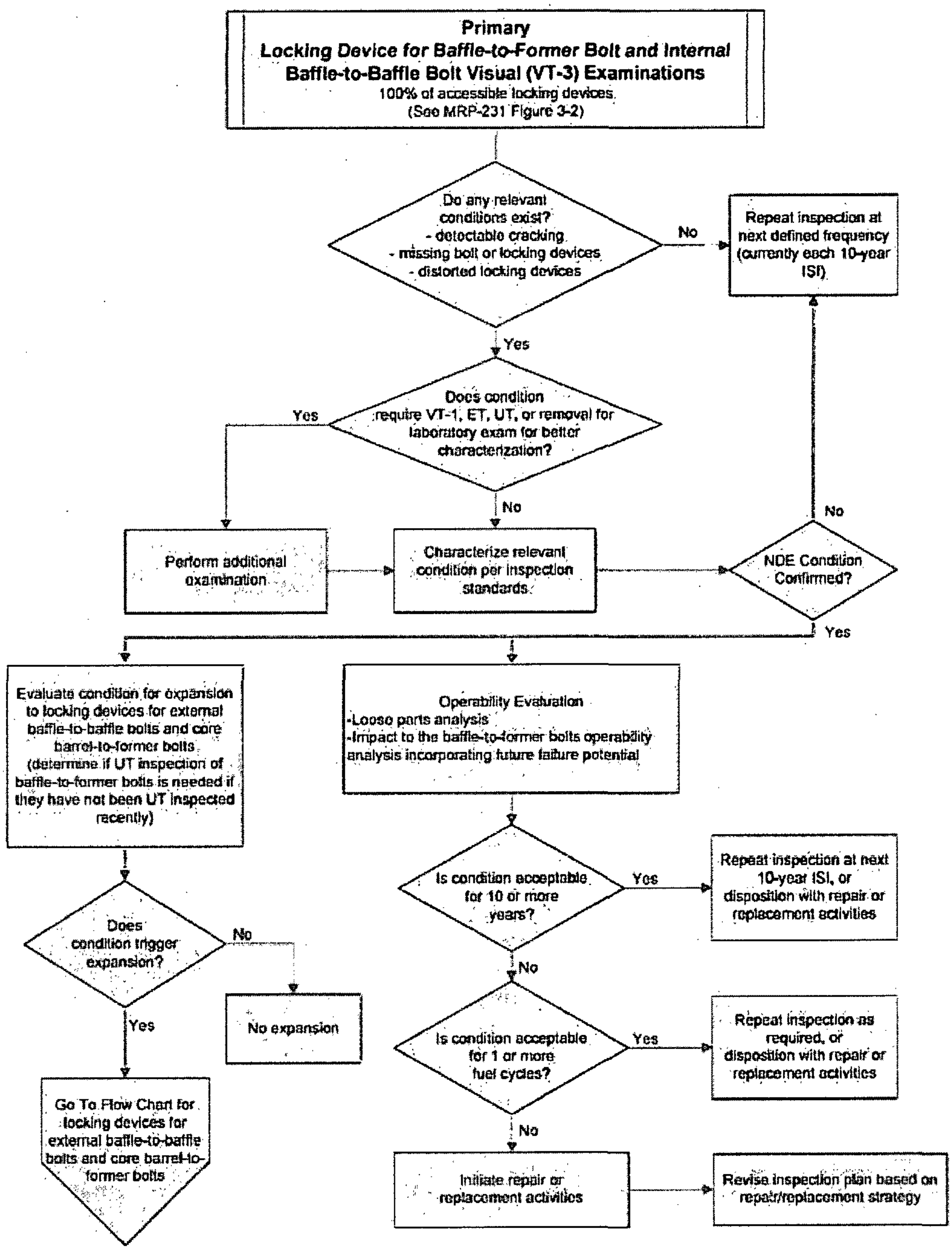
B.1.7 Baffle-to-Former Bolts



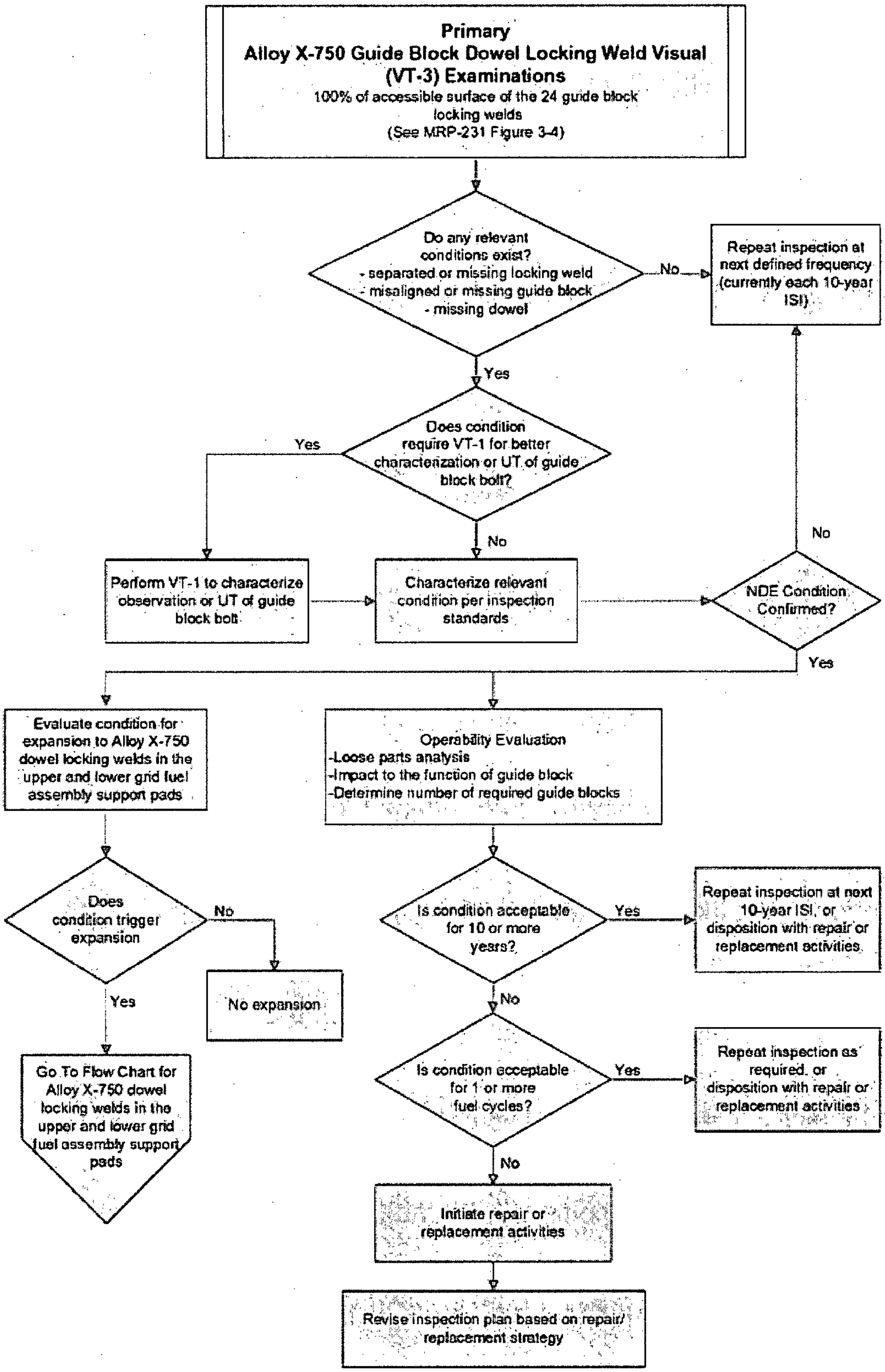
B.1.8 Baffle Plates



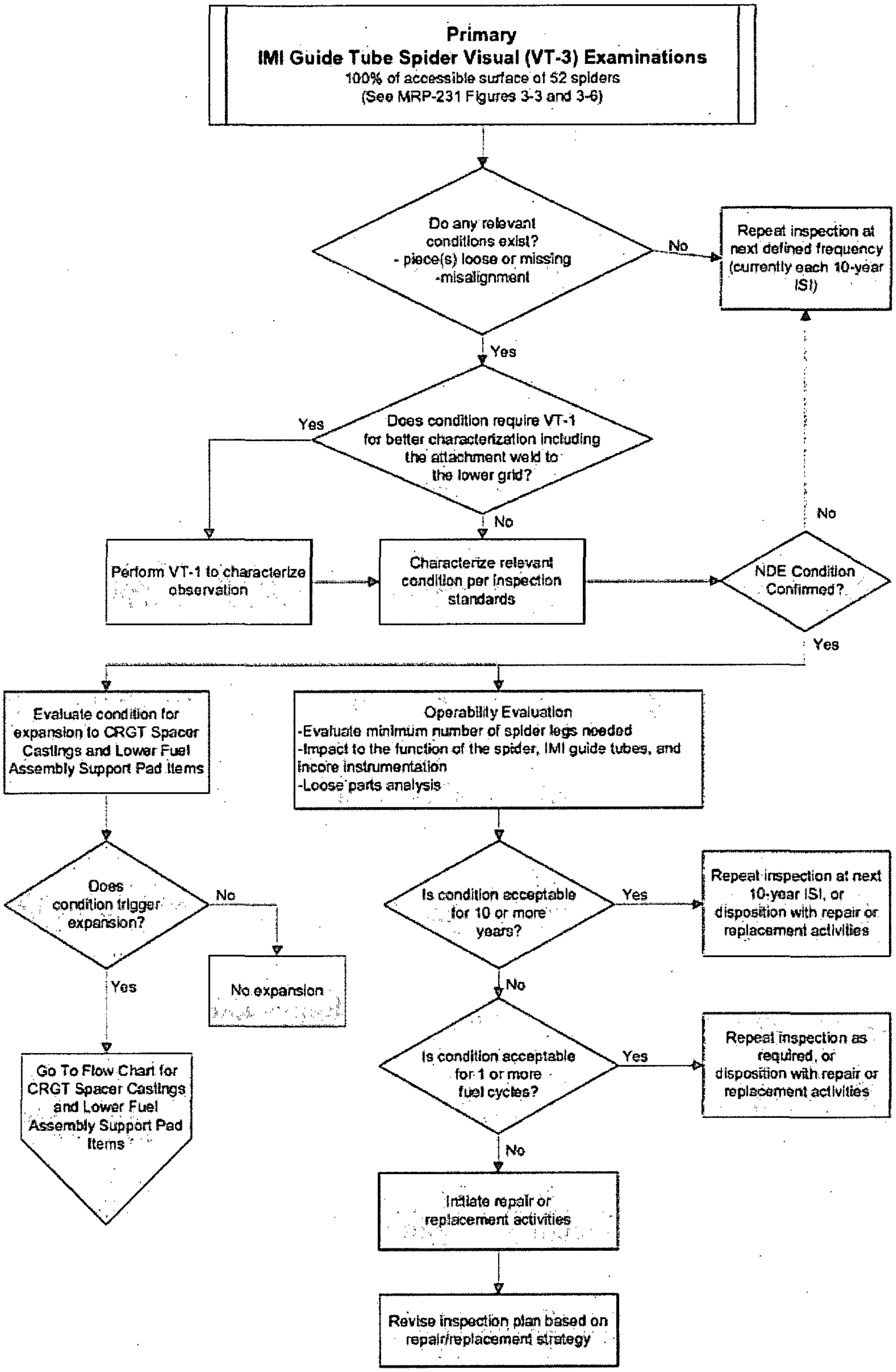
B.1.9 Locking Devices of Baffle-to-Former and Internal Baffle-to-Baffle Bolts



B.1.10 Guide Block Dowel Welds



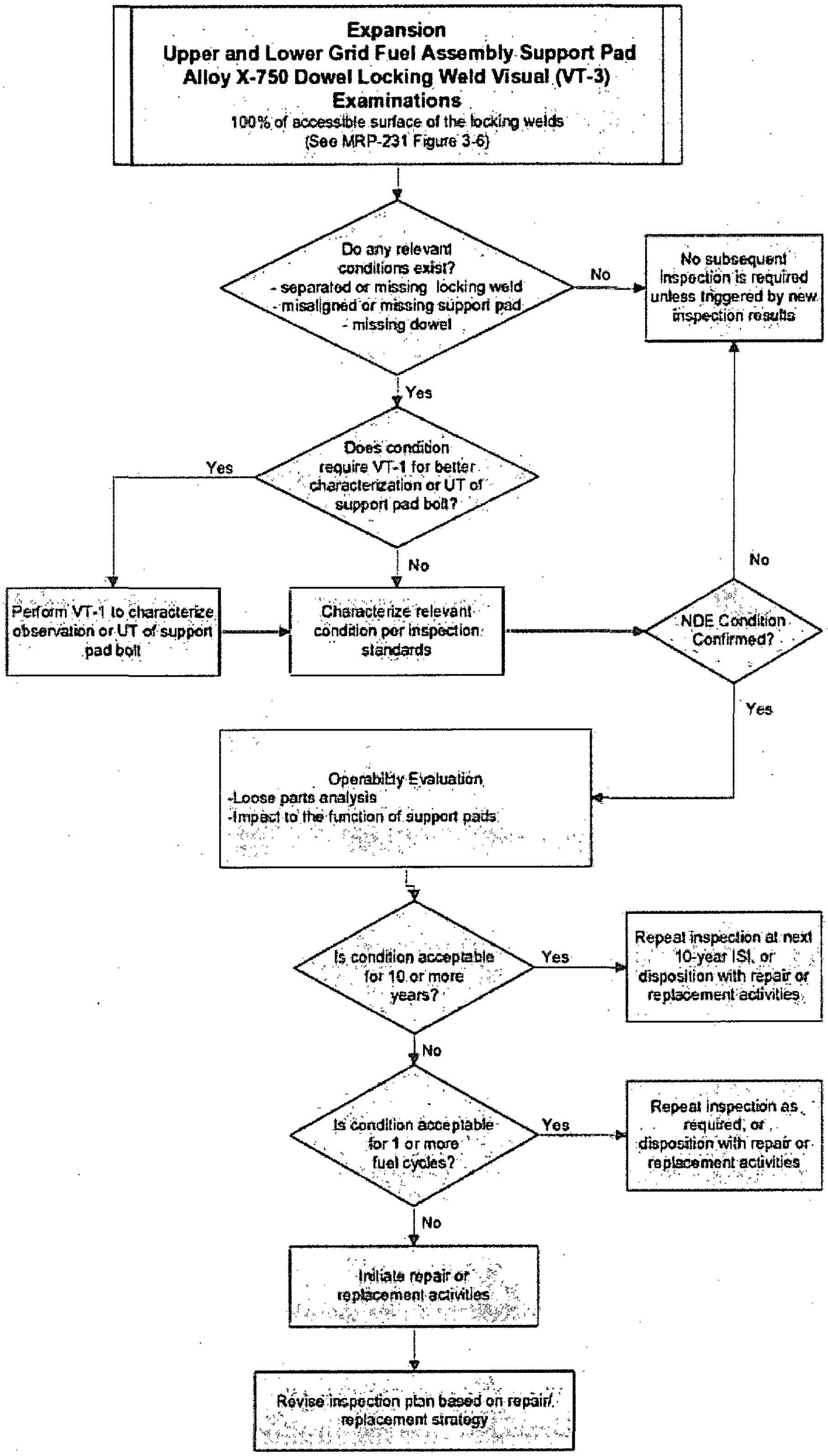
B.1.11 Incore Monitoring Instrumentation Guide Tube Spiders and Welds



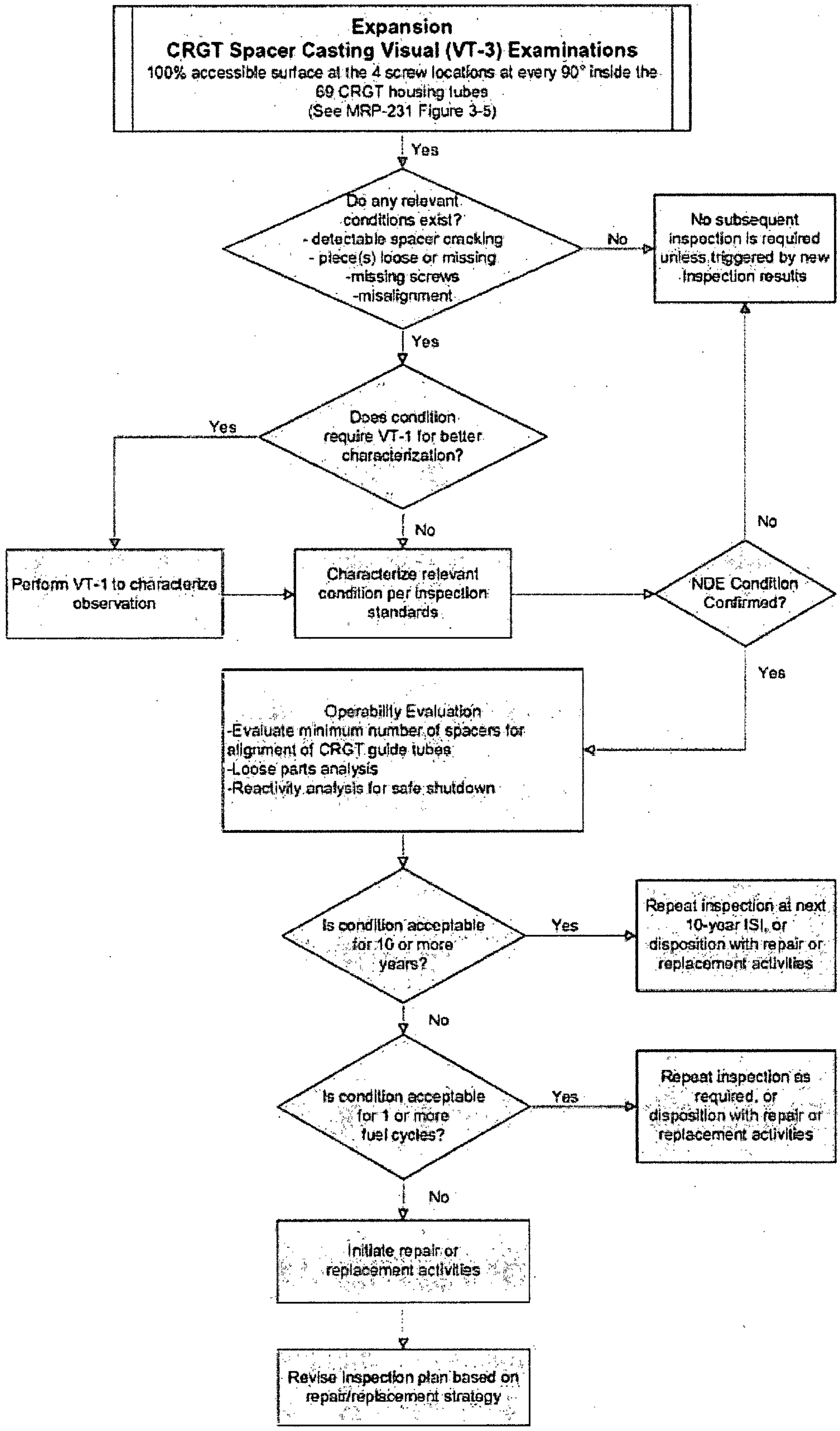
B.2 EXPANSION COMPONENT ITEMS

Logic charts for each of the Expansion component items are provided in this section. A separate sub-section is provided for each component item logic chart.

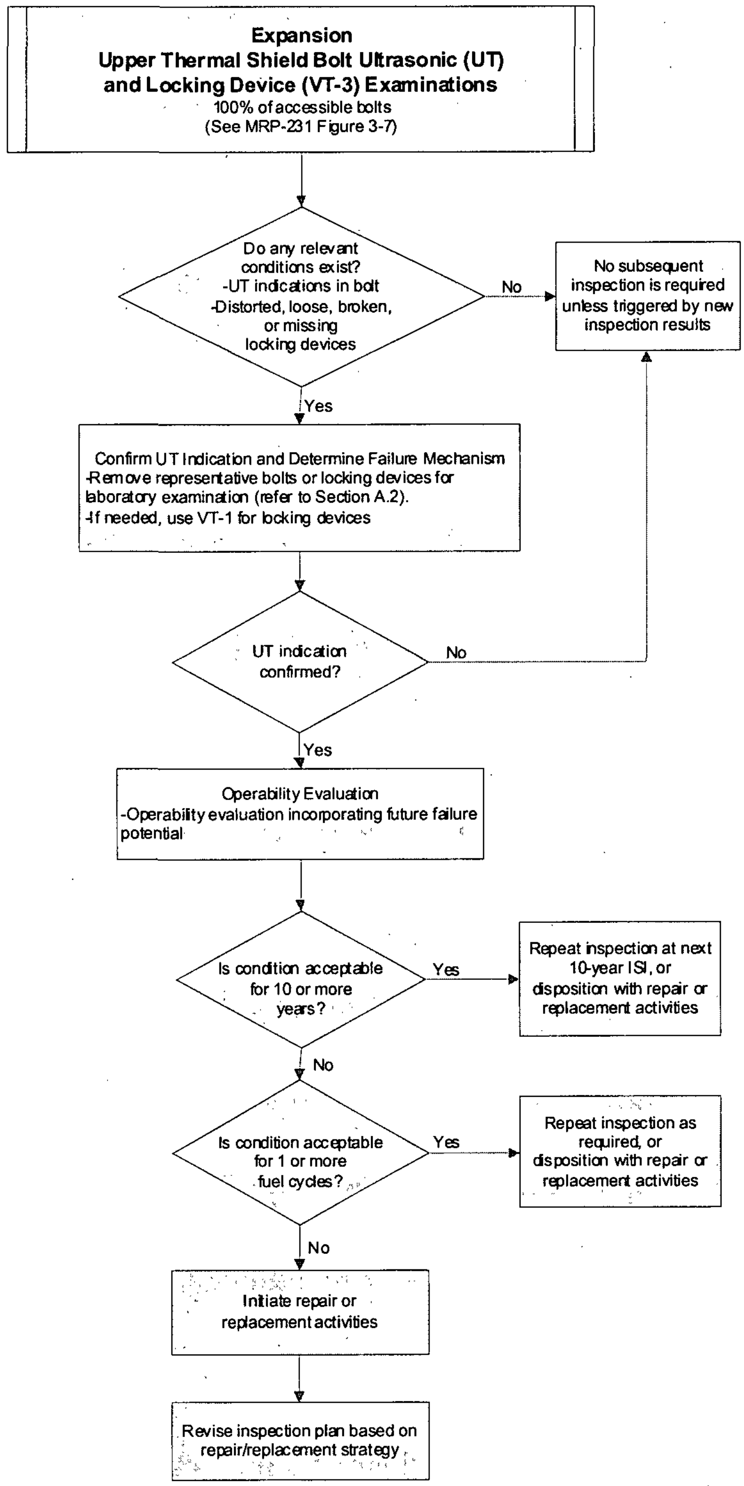
B.2.1 Upper and Lower Grid Fuel Assembly Support Pad Dowel Welds



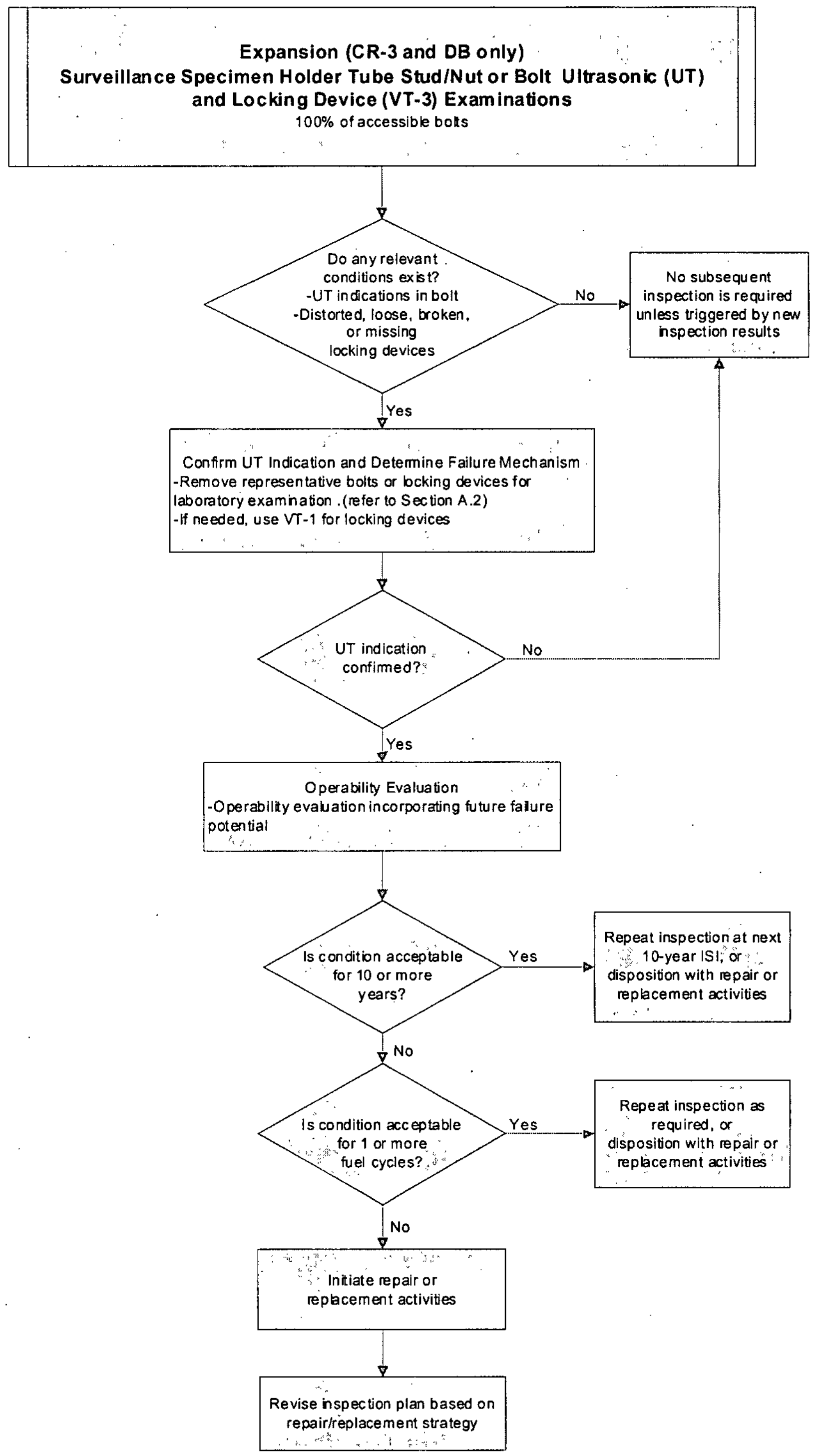
B.2.2 Control Rod Guide Tube Spacer Castings



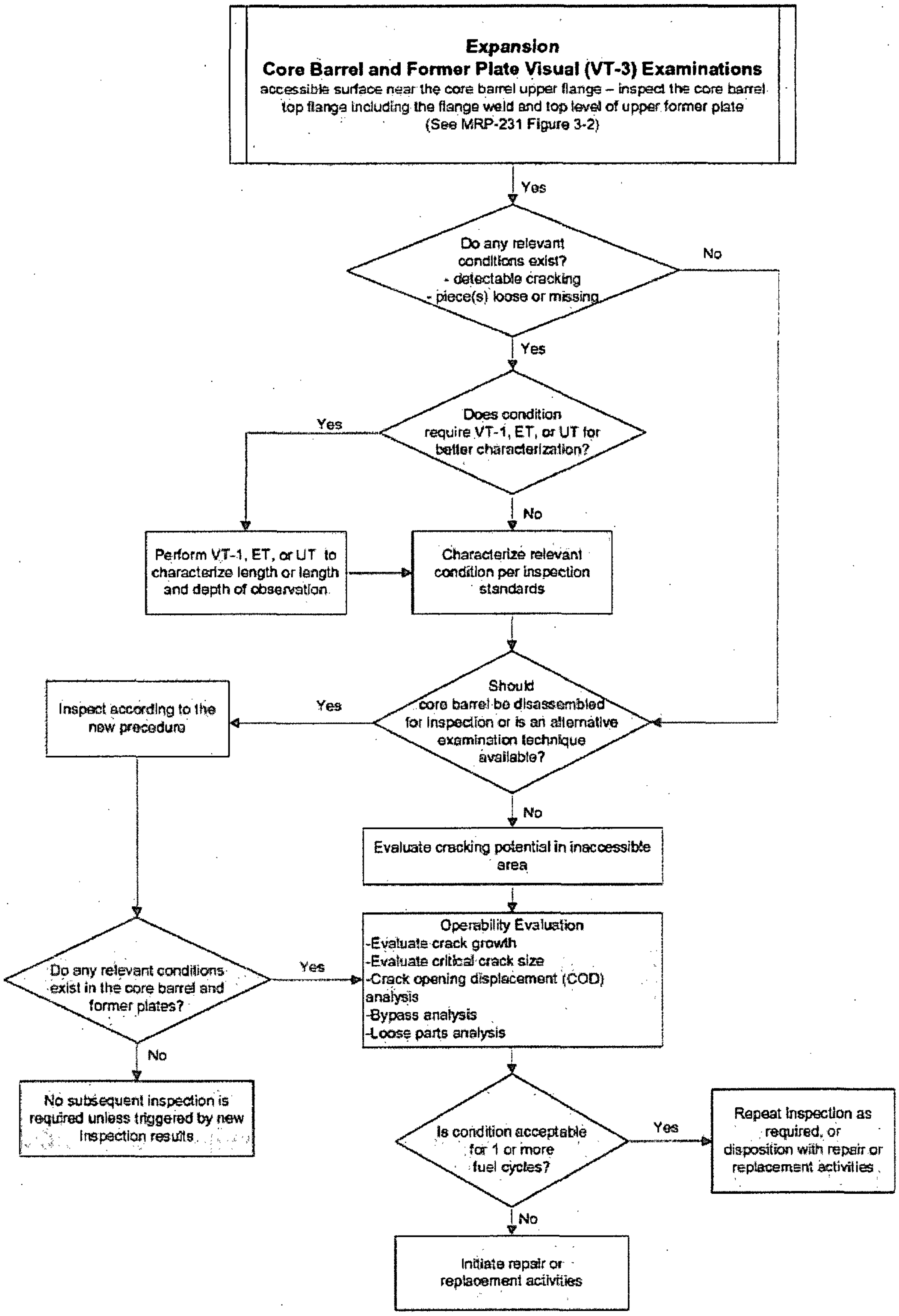
B.2.3 Upper Thermal Shield Bolts and Locking Devices



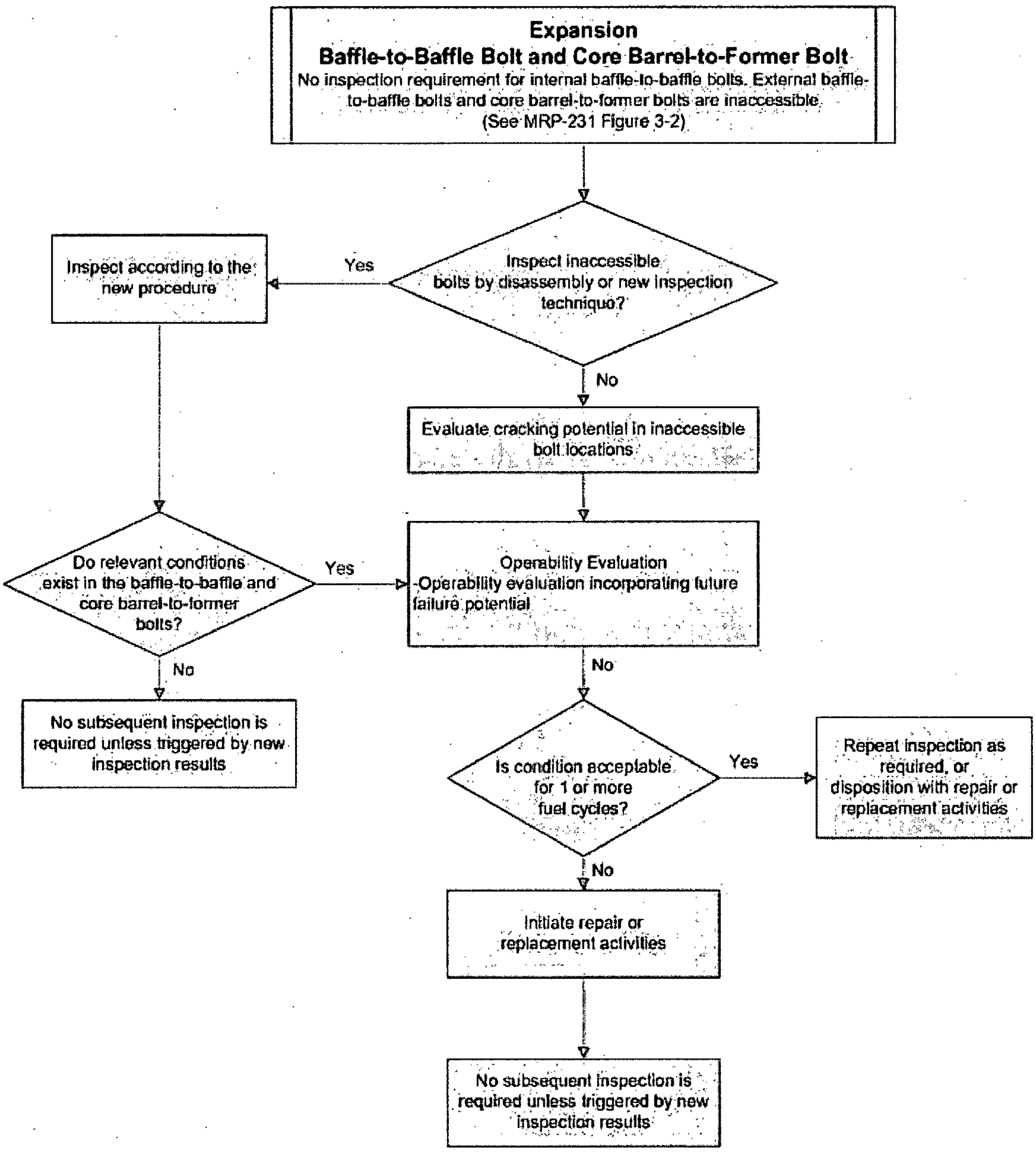
B.2.4 Surveillance Specimen Holder Tube Studs/Nuts or Bolts and Locking Devices



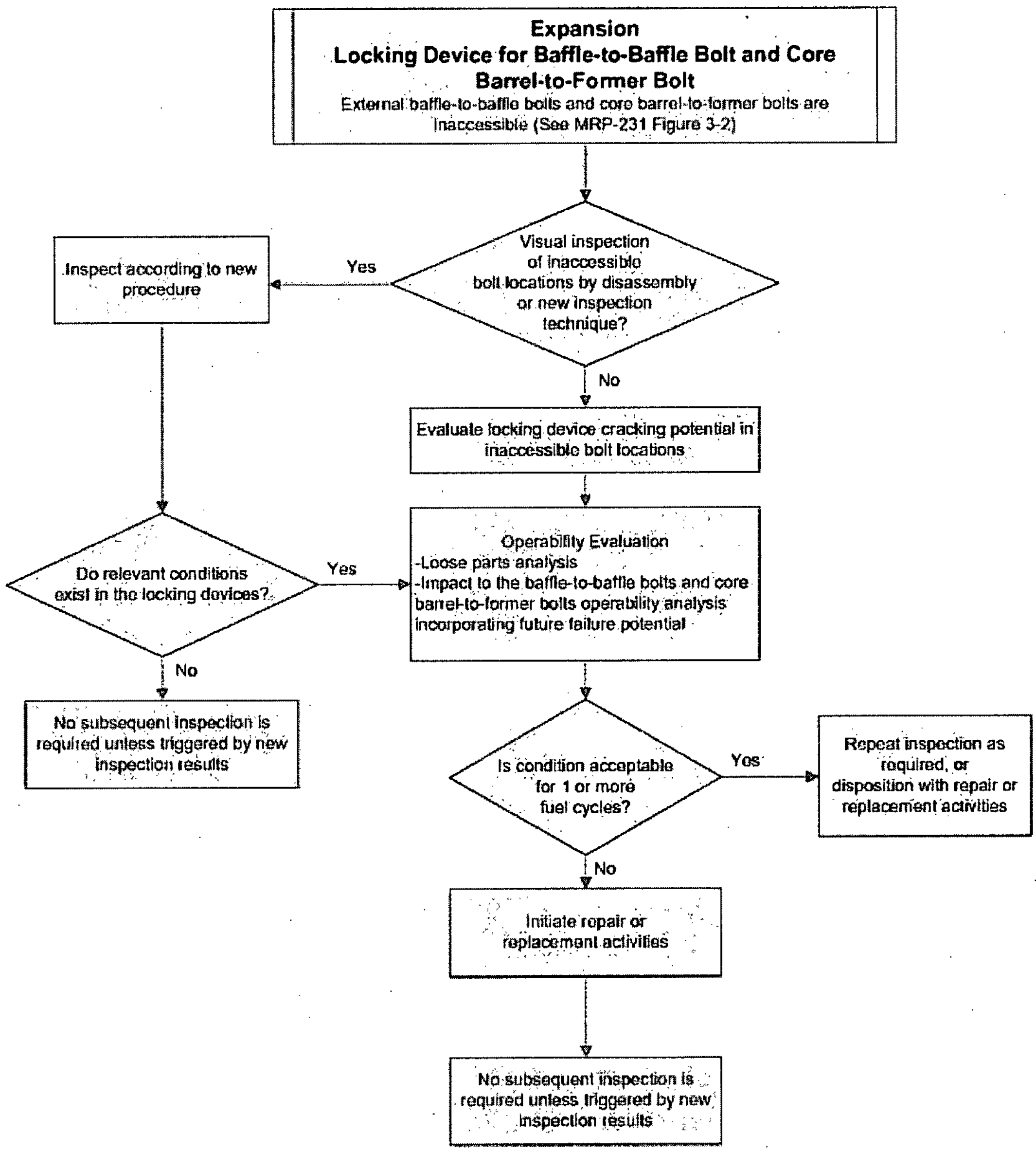
B.2.5 Core Barrel Cylinder and Former Plates



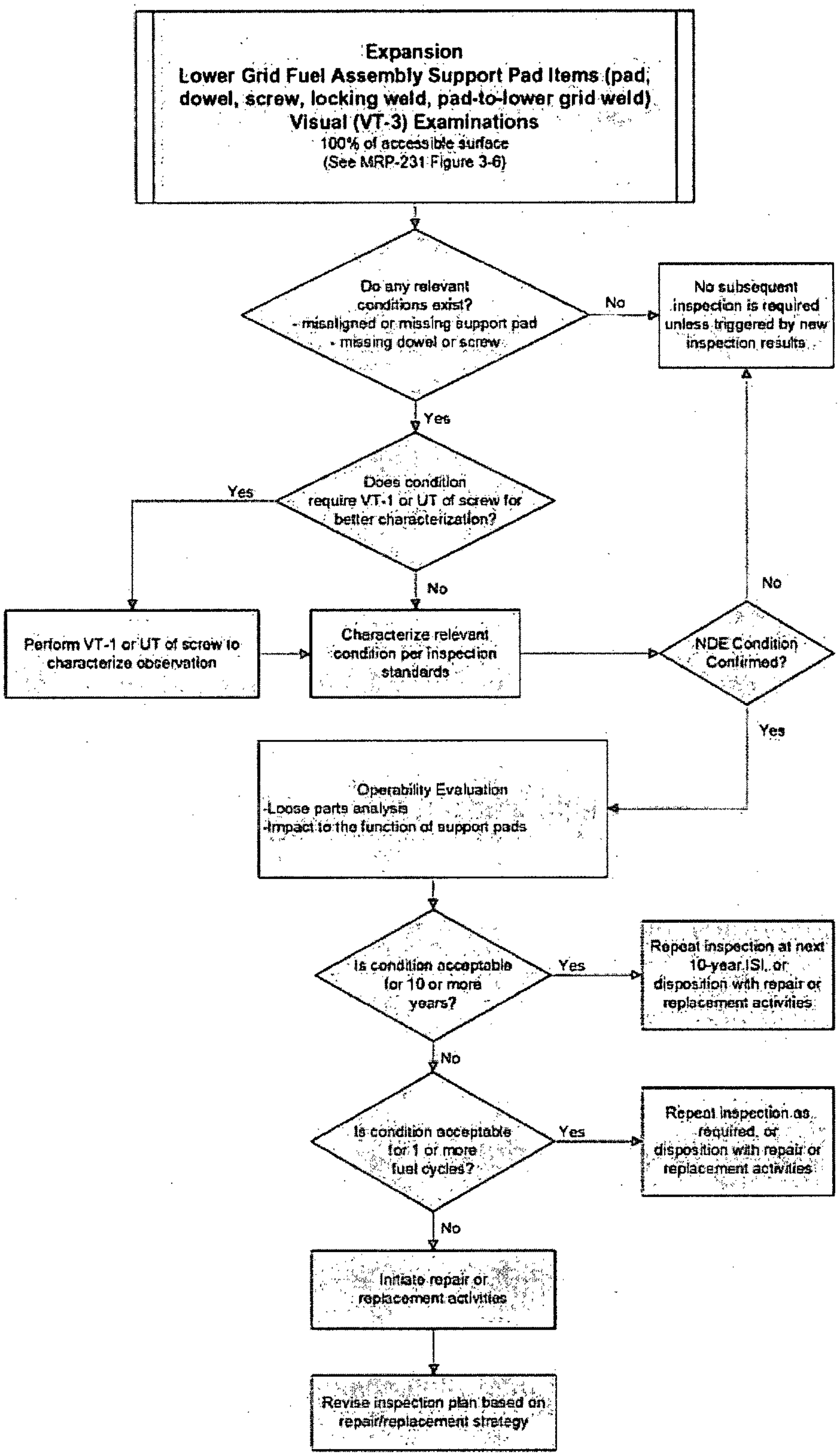
B.2.6 Baffle-to-Baffle Bolts Core Barrel-to-Former Bolts



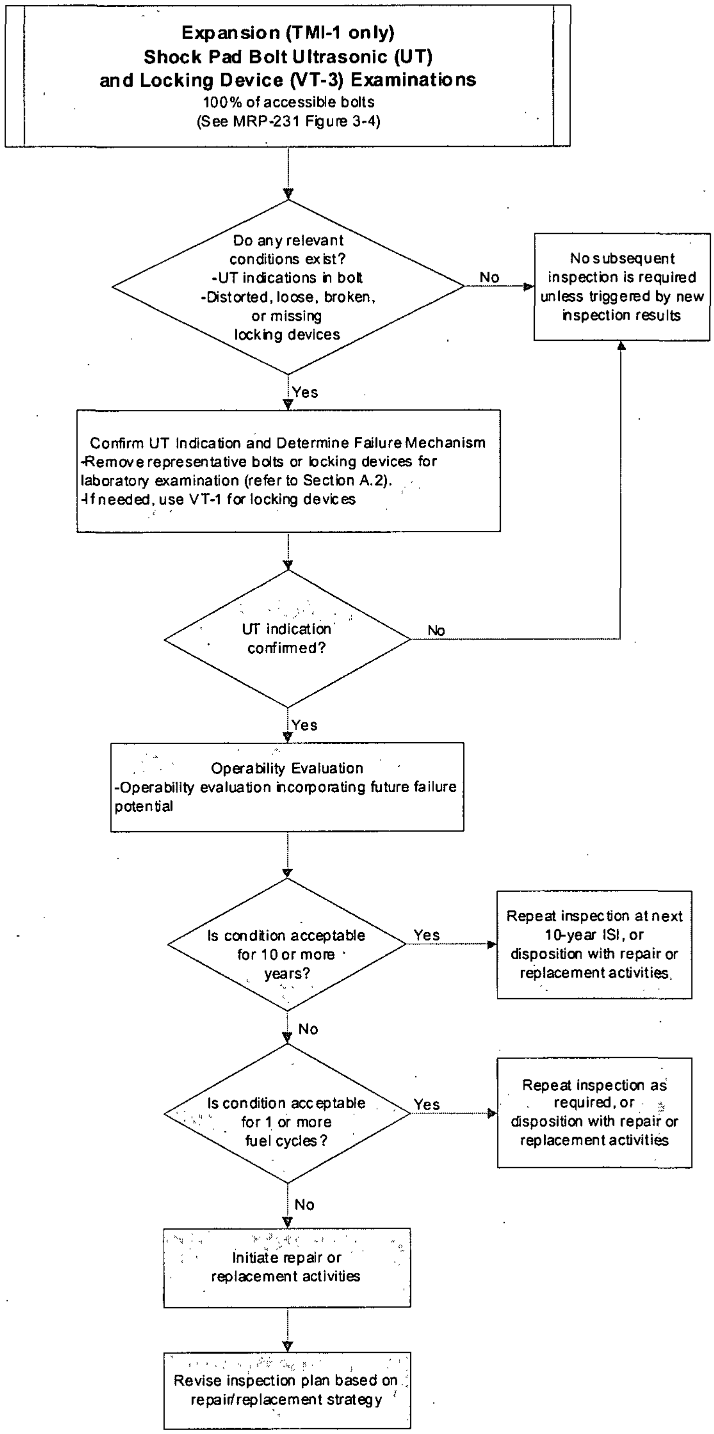
B.2.7 Locking Devices for External Baffle-to-Baffle and Core Barrel-to-Former Bolts



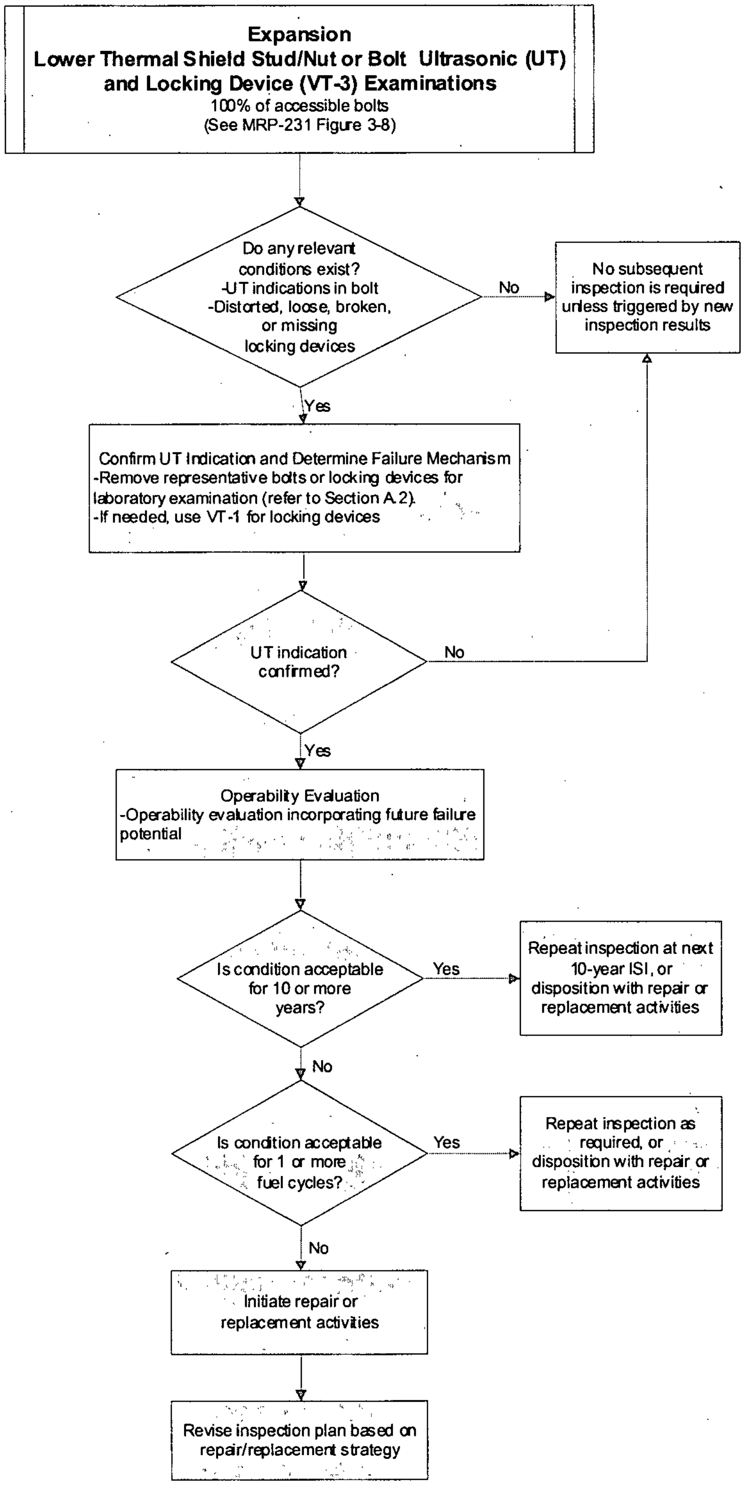
B.2.8 Lower Grid Fuel Assembly Support Pad Items



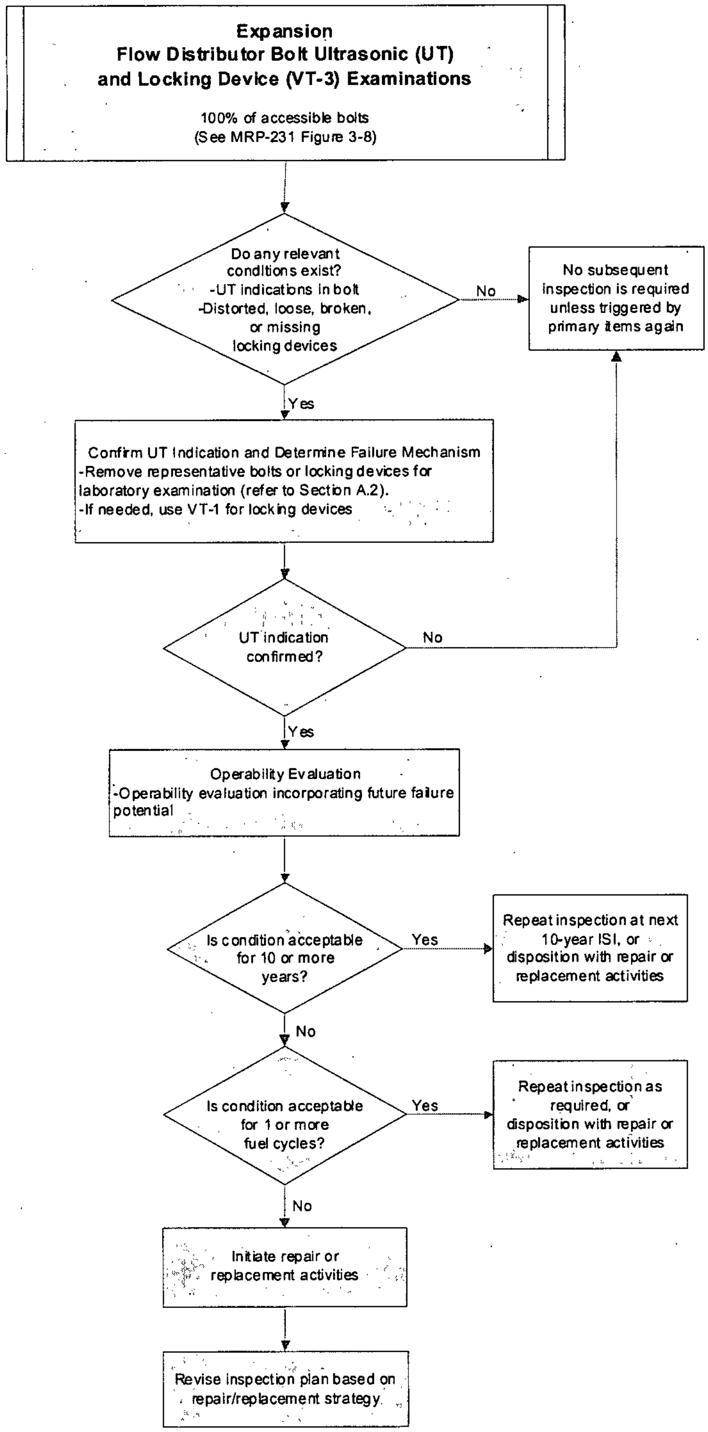
B.2.9 Lower Grid Shock Pad Bolts and Locking Devices



B.2.10 Lower Thermal Shield Bolts and Locking Devices



B.2.11 Flow Distributor Bolts and Locking Devices



APPENDIX C
ACCEPTANCE CRITERIA METHODOLOGY AND DATA REQUIREMENTS FOR COMBUSTION ENGINEERING COMPONENTS INCLUDED IN MRP-227

CE Primary and Expansion Components

CE-ID: 1 Core Shroud Assembly (Bolted) – Core Shroud Bolts

- CE-ID: 1.1 Core Shroud Assembly (Bolted) – Barrel-Shroud Bolts
- CE-ID: 1.2 Core Shroud Assembly (Bolted) – Core Support Column Bolts

CE-ID: 2 Core Shroud Assembly (Welded) – Welds

- CE-ID: 2.1 Core Shroud Assembly (Welded) – Remaining Axial Welds

CE-ID: 3 Core Shroud Assembly (Welded – Full Height) – Shroud Plates

- CE-ID: 3.1 Core Shroud Assembly (Welded – Full Height) – Axial Welds, Ribs and Rings

CE-ID: 4 Core Shroud Assembly (Bolted) – Assembly
CE-ID: 5 Core Shroud Assembly (Welded) – Assembly
CE-ID: 6 Core Support Barrel Assembly – Upper (Core Support Barrel) Flange Weld

- CE-ID: 6.1 Core Support Barrel Assembly – Lower Core Barrel Flange
- CE-ID: 6.2 Core Support Barrel Assembly – Remaining Core Barrel Assembly Welds
- CE-ID: 6.3 Lower Support Structure – Core Support Column Welds

CE-ID: 7 Core Support Barrel Assembly – Lower Flange Weld
CE-ID: 8 Lower Support Structure – Core Support Plate
CE-ID: 9 Upper Internals Assembly – Fuel Alignment Plate
CE-ID: 10 Control Element Assembly – Instrument Guide Tubes

- CE-ID: 10.1 Control Element Assembly – Remaining Instrument Guide Tubes

CE-ID: 11 Lower Support Structure – Deep Beams

CE-ID: 1	**Core Shroud Assembly (Bolted)**		
	Core Shroud Bolts		
Category:	Primary	Applicability:	Bolted plant designs

Degradation Effect:	Cracking (IASCC, fatigue)
Expansion Link:	Core support column bolts, barrel-shroud bolts
Function:	The shroud-former bolts fasten the shroud plates to the barrel-former structure.

Inspection

Method:	Baseline volumetric (UT) examination between 25 and 35 EFPY, with subsequent examination after 10 to 15 additional EFPY to confirm stability of bolting pattern. Re-examination for high-leakage core designs requires continuing inspections on a 10-year interval.
Coverage:	100% of accessible bolts, or as supported by plant-specific justification. Heads are accessible from the core side. UT accessibility may be affected by complexity of head and locking device designs. See MRP-227 Figure 4-24.
Observable Effect:	UT should reliably detect flaws greater than 30% through-shaft cracking.

Failure

Failure Mechanism:	Known IASCC cracking of similar highly irradiated bolts has been reported.
Failure Effect:	Loss of structural stability
Failure Criteria:	Require a minimum bolting pattern

Methodology

Goal:	Must demonstrate that projected number of additional bolt failures will not threaten minimum pattern prior to next scheduled inspection.
Data Requirements:	Loads Bolting patterns Shroud design Fast neutron (dpa) distribution in core shroud Projected bolt failure rate Minimum bolting pattern analysis

CE-ID: 1 **Core Shroud Assembly (Bolted)**

Core Shroud Bolts

Analysis: The observed pattern of failed bolts must meet the pre-defined acceptable bolt pattern and have a reasonable margin to protect against additional failures during the inspection interval. The margin is defined in terms of the number of intact bolts beyond the number required for the minimum bolting pattern. The margin, M, at any time is simply:

$$M = N - Nreq - Nf$$

where

N = total number of shroud-former bolts

Nreq = number of shroud-former bolts in minimum acceptable pattern

Nf = number of failed bolts.

Assuming that there are no failed bolts at the beginning of life, the initial margin is simply: (N - Nreq). For operation through the next 10-15 EFPY interval, require that no more than 50% of initial margin be consumed at the time of the first inspection.

Acceptance Criteria: Procedures for establishing acceptable bolting patterns or the baffle-to-former bolts in Westinghouse-designed plants have been established in [13]. This methodology has been reviewed and accepted by the NRC in a Safety Evaluation in 1998 (TAC No. MA1152). The same methodology should be applied to the two operating CE plants with bolted core shrouds.

1. Observed pattern of unfailed bolts meets pre-defined acceptance criteria.

Approach: No generic effort required. Only two plants are affected

CE-ID: 1.1	**Core Shroud Assembly (Bolted)**		
	Barrel-shroud Bolts		
Category:	Expansion	Applicability:	Bolted plant designs
Degradation Effect:	Cracking (IASCC, fatigue)		
Expansion Link:	Core shroud bolts		
Function:	Maintain structural integrity of barrel-shroud structure.		

Inspection

Method:	Volumetric (UT) examination, with initial and subsequent examination frequencies dependent on the results of core shroud bolt examinations.
Coverage:	100% (or as supported by plant-specific justification) of barrel-shroud and guide lug insert bolts with neutron fluence exposures > 3 displacements per atom (dpa).
Observable Effect	UT should reliably detect flaws greater than 30% through-shaft cracking.

Failure

Failure Mechanism:	Cracking by combined effects of IASCC and fatigue.
Failure Effect:	Inability to maintain structural stability
Failure Criteria:	Require a minimum bolting pattern.

Methodology

Goal: Must demonstrate a minimum bolting pattern.

Data Requirements:
- Bolting patterns
- Shroud design
- Fast neutron (dpa) distribution in core shroud
- Projected bolt failure rate
- Minimum bolting pattern analysis

Analysis: The observed pattern of failed bolts must meet the pre-defined acceptable bolt pattern and have a reasonable margin to protect against additional failures during the inspection interval. The margin is defined in terms of the number of intact bolts beyond the number required for the minimum bolting pattern. The margin, M, at any time is simply:

$$M = N - Nreq - Nf$$

where

N = total number of barrel-former bolts

Nreq = number of barrel-former bolts in minimum acceptable pattern

Nf = number of failed bolts.

Assuming that there are no failed bolts at the beginning of life, the initial margin is simply: (N-Nreq). For operation through the next 10-15 EFPY interval, require that no more than 50% of initial margin be consumed at the time of the first inspection.

CE-ID: 1.1 Core Shroud Assembly (Bolted)

Barrel-shroud Bolts

Acceptance Criteria: Procedures for establishing acceptable bolting patterns or the barrel-to-former bolts in Westinghouse designed plants have been established in [13]. This methodology has been reviewed and accepted by the NRC in a Safety Evaluation in 1998 (TAC No. MA1152). The same methodology should be applied to the two operating CE plants with bolted core shrouds.

1. Observed pattern of unfailed bolts meets pre-defined acceptance criteria.

Approach: No generic effort required. Only two plants are affected

CE-ID: 1.2	**Core Shroud Assembly (Bolted)**		
	Core support column bolts		
Category:	Expansion	Applicability:	Bolted plant designs
Degradation Effect:	Cracking (IASCC, fatigue)		
Expansion Link:	Core shroud bolts		
Function:	Attach core support columns to core support plate.		

Inspection

Method:	Ultrasonic (UT) examination, with initial and subsequent examination frequencies dependent on the results of core shroud bolt examinations.
Coverage:	100% (or as supported by plant-specific analysis) of core support column bolts with neutron fluence exposures > 3 dpa.
Observable Effect:	UT should reliably detect flaws greater than 30% through-shaft cracking.

Failure

Failure Mechanism:	IASCC and fatigue
Failure Effect:	Loss of structural stability
Failure Criteria:	Determine minimum number of support columns required to maintain structural integrity.

Methodology

Goal: Establish functional requirements for core support columns.

- During normal operation system of support columns should resist core plate deformation due to mechanical or thermal loading. Core plate requirements for "flatness" and fuel assembly alignment.
- During limiting accident transient system must maintain structural integrity.

Data Requirements: Loads on core support plate.

Displacement tolerances on lower core plate.

Analysis: See MRP-227 Figures 4-16 and 4-33. Build FEA model of lower support structure that includes support columns and core support plate. Model should be capable of removing individual column or breaking attachment to lower core support plate. Would require multiple iterations to establish "minimum acceptable patterns" of core support columns and support column bolts.

Structural model must be run for functional requirements A and B.

Determine margin for additional failures.

1. Assume number of failures in next 10 years is equal to number observed to date.

   N = # of support columns

   Nf = # of observed flawed columns

   Nreq = # of columns in relevant minimum pattern

   Margin = N-Nreq

CE-ID: 1.2 **Core Shroud Assembly (Bolted)**

Core support column bolts

Acceptance Criteria: Require that no more of 1/2 of columns in margin are failed:

$$Nf < (N\text{-}Nreq)/2$$

Approach: Generic program to share first-of-a-kind effort. (See W-ID: 2-1)

- Pilot analysis of lower support structure to identify critical issues.
- Expect final acceptance based on plant-specific analysis.

CE-ID: 2 Core Shroud Assembly (Welded)

Category:	Primary	Applicability:	Plant designs with core shrouds assembled in two vertical sections

Degradation Effect: Cracking (IASCC)

Expansion Link: Remaining axial welds

Function:
1. Maintain core geometry.
2. Direct coolant flow.

Inspection

Method: Enhanced visual (EVT-1) examination no later than 2 refueling outages from the beginning of the license renewal period and subsequent examination on a 10-year interval.

Coverage: Axial and horizontal weld seams at the core shroud re-entrant corners as visible from the core side of the shroud, within six inches of central flange and horizontal stiffeners.

See MRP-227 Figures 4-12 and 4-14.

Observable Effect: Cracking

Failure

Failure Mechanism: SCC

Failure Effect:
1. Core damage caused by event. Require maintenance of coolable geometry.
2. Damage to peripheral fuel assemblies.
3. Through-wall crack provides leak path through shroud.

Failure Criteria: Observed flaw will not grow to critical crack size for crack initiation during limiting transient event prior to next planned inspection.

No observable damage in corresponding sections of peripheral fuel assemblies.

Methodology

Goal: Perform flaw-tolerance analysis to demonstrate that crack will not grow to exceed crack initiation size limit during limiting transient events.

Data Requirements:
1. Normal operating loads (plant specific)
2. Elastic-plastic K solution for normal operation (geometry dependent)
3. Fast neutron fluence (or dpa) at crack location (plant specific)
4. IASCC crack growth rate curve
5. Limiting transient loads (plant specific)
6. K solution for limiting transient (geometry dependent)
7. Irradiated fracture toughness (K_{Ic})

CE-ID: 2 Core Shroud Assembly (Welded)

Analysis:

1. Assume through-wall crack of length (L) corresponding to visual indication.
2. Estimate normal operating loads at crack tip as determined by:
   - Weld residual stresses
   - Irradiation induced stress relaxation
   - Swelling induced stresses
   - Normal operation (Delta-P, Delta-T, flow, dead weight)

   Note: May be reasonable to assume that residual stress and stress relaxation are offsetting factors.
3. Obtain stress intensity factor (K) solution corresponding to crack at corner with described loads.
4. Construct models for fatigue and IASCC crack growth rates as a function of K.
5. Integrate crack growth rate over next inspection interval to estimate crack length. Note: This may be accomplished numerically.
6. Estimate limiting transient load (presumably LOCA).
7. Obtain stress intensity factor (K_{app}) solution corresponding to crack at corner with transient loads.
8. For center of core shroud location, use limiting fracture toughness, K_{Ic}, for highly irradiated material.

Acceptance Criteria: Structure is acceptable if $K_{app} < K_{Ic}$

Approach: Expect calculation to be plant specific

- Define general load conditions at weld seams.
- Define K-solution for loading at weld seams.

CE-ID: 2.1	**Core Shroud Assembly (Welded)**		
	Remaining Axial Welds		
Category:	Expansion	Applicability:	Plant designs with core shrouds assembled in two vertical sections

Degradation Effect: Cracking (IASCC)

Expansion Link: Core shroud plate-former plate weld.

Function:
1. Maintain core geometry.
2. Direct coolant flow.

Inspection

Method: Enhanced visual (EVT-1) examination, with initial and subsequent examination frequencies dependent on the results of the core shroud weld examinations.

Coverage: Axial weld seams other than the core shroud re-entrant corner welds at the core mid-plane.

Observable Effect: Cracking

Failure

Failure Mechanism: IASCC

Failure Effect:
1. Core damage caused by event – require maintenance of coolable geometry.
2. Damage to peripheral fuel assemblies.
3. Through-wall crack provides leak path through shroud.

Failure Criteria:
1. Observed flaw will not grow to critical crack size for crack initiation during limiting transient event prior to next planned inspection.
2. No observable damage in corresponding sections of peripheral fuel assemblies

Methodology

Goal: Demonstrate that cracks in axial welds are stable.

Data Requirements:
1. Normal operating loads (plant specific)
2. Elastic-plastic K solution for normal operation (geometry dependent)
3. Fast neutron fluence (or dpa) at crack location (plant specific)
4. IASCC crack growth rate curve
5. Limiting transient (potentially LOCA) loads (plant specific)
6. K solution for limiting transient (geometry dependent)
7. Irradiated fracture toughness (K_{Ic})

CE-ID: 2.1 Core Shroud Assembly (Welded)

Remaining Axial Welds

Analysis:

1. Assume through-wall crack of length (L) corresponding to visual indication.
2. Estimate normal operating loads at crack tip as determined by:
   - Weld residual stresses
   - Irradiation induced stress relaxation
   - Swelling induced stresses
   - Normal operation (Delta-P, Delta-T, flow, dead weight)

Note: May be reasonable to assume that residual stress and stress relaxation are offsetting factors.

3. Obtain stress intensity factor (K) solution corresponding to crack at corner with described loads.
4. Construct models for fatigue and IASCC crack growth rates as a function of K.
5. Integrate crack growth rate over next inspection interval to estimate crack length. Note: This may be accomplished numerically.
6. Estimate limiting transient load (presumably LOCA).
7. Obtain stress intensity factor (K_{app}) solution corresponding to crack at corner with transient loads.
8. For center of core shroud location, use limiting fracture toughness, K_{Ic}, for highly irradiated material.

Acceptance Criteria: Structure is acceptable if $K_{app} < K_{Ic}$

Approach: Plant-specific analysis.

- Require flaw tolerance handbook/methodology based on flaw location and direction.

CE-ID: 3 **Core Shroud Assembly (Welded)**

Shroud Plates

Category: Primary Applicability: Plant designs with core shrouds assembled with full-height shroud plates

Degradation Effect: Cracking (IASCC)

Expansion Link: Remaining axial welds, ribs and rings

Function:
1. Maintain core geometry.
2. Direct coolant flow.

Inspection

Method: Enhanced visual (EVT-1) examination no later than 2 refueling outages from the beginning of the license renewal period and subsequent examination on a 10-year interval.

Coverage: Axial weld seams at the core shroud re-entrant corners, at the core mid-plane (±three feet in height) as visible from the core side of the shroud. See MRP-227, Figure 4-13.

Observable Effect: Cracking

Failure

Failure Mechanism: IASCC

Failure Effect:
1. Core damage caused by event – require maintenance of coolable geometry.
2. Damage to peripheral fuel assemblies.
3. Through-wall crack provides leak path through shroud.

Failure Criteria:
1. Observed flaw will not grow to critical crack size for crack initiation during transient loading condition event prior to next planned inspection.
2. No observable damage in corresponding sections of peripheral fuel assemblies.

Methodology

Goal: Perform flaw-tolerance analysis to demonstrate that crack will not grow to exceed crack initiation size limit during limiting transient events.

Data Requirements:
1. Normal operating loads (plant specific)
2. Elastic-plastic K solution for normal operation (geometry dependent)
3. Fast neutron fluence (or dpa) at crack location (plant specific)
4. IASCC crack growth rate curve
5. Limiting transient loads (plant specific)
6. K solution for limiting transient (geometry dependent)
7. Irradiated fracture toughness (K_{Ic})

CE-ID: 3 — Core Shroud Assembly (Welded)

Shroud Plates

Analysis:

1. Assume through-wall crack of length (L) corresponding to visual indication.
2. Estimate normal operating loads at crack tip as determined by:
   - Weld residual stresses
   - Irradiation induced stress relaxation
   - Swelling induced stresses
   - Normal operation (Delta-P, Delta-T, flow, dead weight)

Note: May be reasonable to assume that residual stress and stress relaxation are offsetting factors.

3. Obtain stress intensity factor (K) solution corresponding to crack at corner with described loads.
4. Construct models for fatigue and IASCC crack growth rates as a function of K.
5. Integrate crack growth rate over next inspection interval to estimate crack length. Note: This may be accomplished numerically.
6. Estimate limiting transient load (presumably LOCA).
7. Obtain stress intensity factor (K_{app}) solution corresponding to crack at corner with transient loads.
8. For center of core shroud location, use limiting fracture toughness, K_{Ic}, for highly irradiated material.

Acceptance Criteria: Structure is acceptable if $K_{app} < K_{Ic}$.

Approach: No generic analysis: Only one utility with this design.

CE-ID: 3.1	**Core Shroud Assembly (Welded)** **Remaining Axial Welds, Ribs and Rings**		
Category:	Expansion	Applicability:	Plant designs with core shrouds assembled with full-height shroud plates

Degradation Effect: Cracking (IASCC)

Expansion Link: Shroud plates of welded core shroud assemblies

Function: 1. Maintain dimensional stability of core shroud plus ribs and rings.

Inspection

Method: Enhanced visual (EVT-1) examination, with initial and subsequent examination frequencies dependent on the results of the core shroud weld examinations.

Coverage: Axial weld seams other than the core shroud re-entrant corner welds at the core mid-plane.

Observable Effect: Cracking

Failure

Failure Mechanism: IASCC

Failure Effect:
1. Deformation of core barrel leads to interaction with fuel.
2. Unable to withstand limiting transient loads due to a loss of structural support.
3. Generation of loose parts.

Failure Criteria: Welds with observable cracks assumed failed.

Components with cracks in all attachment welds considered as potential loose part.

Require minimum acceptable support structure to withstand limiting transient forces.

Methodology

Goal: Demonstrate that:
1. Cracks in axial welds are stable
2. Cracks in ribs and rings will not generate loose parts

Data Requirements:
1. Normal operating loads (plant specific)
2. Elastic-plastic K solution for normal operation (geometry dependent)
3. Fast neutron fluence (or dpa) at crack location (plant specific)
4. IASCC crack growth rate curve
5. Limiting transient loads (plant specific)
6. K solution for limiting transient (geometry dependent)
7. Irradiated fracture toughness (K_{Ic})

Analysis: For remaining axial welds:
1. Assume through-wall crack of length (L) corresponding to visual indication.

CE-ID: 3.1 **Core Shroud Assembly (Welded)**

Remaining Axial Welds, Ribs and Rings

2. Estimate normal operating loads at crack tip as determined by:
   - Weld residual stresses
   - Irradiation induced stress relaxation
   - Swelling induced stresses
   - Normal operation (Delta-P, Delta-T, flow, dead weight)

Note: May be reasonable to assume that residual stress and stress relaxation are offsetting factors.

3. Obtain stress intensity factor (K) solution corresponding to crack at corner with described loads.
4. Construct models for fatigue and IASCC crack growth rates as a function of K.
5. Integrate crack growth rate over next inspection interval to estimate crack length. Note: This may be accomplished numerically.
6. Estimate limiting transient load (presumably LOCA).
7. Obtain stress intensity factor (K_{app}) solution corresponding to crack at corner with transient loads.
8. For center of core shroud location, use limiting fracture toughness, K_{Ic}, for highly irradiated material.
9. Structure is acceptable if $K_{app} < K_{Ic}$.

For ribs and rings:

Prepare for examination by conducting a failure modes and effects analysis (FMEA) to identify full range of potential relevant observations prior to inspection. Primary concern:

- Fracture of weld between shroud and ring
- Fracture of welds in stiffeners

The major effects of these failure mechanisms are expected to be:

- Loss of stability in shroud structure (possible deformation and interaction with fuel assembly)
- Loose parts

Acceptance Criteria: For axial welds: $K_{app} < K_{Ic}$

A plant-specific plan should be developed for evaluating and mitigating the potential relevant conditions related to weld failures in ribs, rings or stiffeners. The evaluation should consider any previously reported observations.

Approach: No generic analysis: Only one utility with this design.

CE-ID: 4	**Core Shroud Assembly (Bolted) Assembly**		
Category:	Primary	Applicability:	Bolted plan designs
Degradation Effect:	Distortion (Void Swelling)		
Expansion Link:	None		
Function:	Provide support, guidance, and protection for the reactor core. Provide a passageway for the distribution of the reactor coolant flow to the reactor core. Provide gamma and neutron shielding for the reactor vessel.		

Inspection

Method:	Visual (VT-3) examination no later than 2 refueling outages from the beginning of the license renewal period. Subsequent examinations on a 10-year interval.
Coverage:	Core side surfaces as indicated. See Figures 4-25 and 4-26 of MRP-227.
Observable Effect:	Degradation of general condition as described above.

Failure

Failure Mechanism:	Void swelling
Failure Effect:	1. Interference with fuel assemblies 2. Obstruction of coolant flow 3. Loose parts generation 4. Distortion/misalignment of core 5. Local temperature peaks 6. Degradation of control rod insertability 7. Baffle jetting
Failure Criteria:	No relevant observations

Methodology

Goal:	A plant-specific plan should be developed for evaluating and mitigating the potential relevant conditions. The evaluation should consider any previously reported observations.
Data Requirements:	Baseline data on previous visual examinations of core shroud. Performance records for peripheral fuel assemblies. Loose parts monitoring data.
Analysis:	Prepare for examination by conducting a failure modes and effects analysis (FMEA) to identify full range of potential relevant observations prior to inspection. Failure mechanisms considered should include: • Broken or missing locking devices, • Protruding bolt heads • Missing bolts or bolt heads.

CE-ID: 4 **Core Shroud Assembly (Bolted)**

Assembly

- Distortion or displacement of shroud plates
- Gross cracking of shroud plates
- Gaps at plate joints
- Interaction with fuel assemblies

Acceptance Criteria: Determined by FMEA

Approach: FMEA should address plant-specific practices and priorities. Some generic work possible to outline issues and options to be addressed in FMEA.

CE-ID: 5 **Core Shroud Assembly (Welded) Assembly**

Category:	Primary	Applicability:	Plant designs with core shrouds assembled in two vertical sections

Degradation Effect: Distortion (void swelling), as evidenced by separation between the upper and lower flanges.

Expansion Link: None

Function: Provide support, guidance, and protection for the reactor core.

Provide a passageway for the distribution of the reactor coolant flow to the reactor core.

Provide gamma and neutron shielding for the reactor vessel.

Inspection

Method: Visual (VT-1) examination no later than 2 refueling outages from the beginning of the license renewal period. Subsequent examinations on a 10-year interval.

Coverage: If a gap exists, make three to five measurements of gap opening from the core side at the core shroud re-entrant corners. Then, evaluate the swelling on a plant-specific basis to determine frequency and method for additional examinations. See MRP-227 Figures 4-12 and 4-14.

Observable Effect: Seam between upper and lower sections should appear even and consistent with any historical records.

Evidence of gaping between plates at protruding corners should be considered a relevant condition.

Failure

Failure Mechanism: Void Swelling

Failure Effect:

1. Potential leakage through shroud.
2. Significant distortion may interfere with peripheral fuel assemblies.
3. Condition is a precursor to high stresses and potential cracking at weld seams.

Failure Criteria:

1. Damage on corresponding peripheral fuel assemblies.
2. Gap size implies peak shroud swelling > 5% by volume.

Methodology

Goal: A plant-specific plan should be developed for evaluating and mitigating the potential relevant conditions. The evaluation should consider any previously reported observations.

Data Requirements:

1. Gap size
2. Swelling deformation model of shroud
3. Shroud fluence distribution
4. Shroud temperature distribution

CE-ID: 5 Core Shroud Assembly (Welded)

Assembly

Analysis:

Prepare for examination by conducting a failure modes and effects analysis (FMEA) to identify full range of potential relevant observations prior to inspection. Failure mechanisms considered should include:

- Broken or missing locking devices
- Protruding bolt heads
- Missing bolts or bolt heads
- Distortion or displacement of shroud plates
- Gross cracking of shroud plates
- Gaps at plate joints
- Interaction with fuel assemblies

Acceptance Criteria:

Quantitative evaluation of swelling would require a time dependent structural model that incorporates the effects of void swelling. Temperature gradients caused by gamma heating must be accurately estimated to provide reliable swelling estimates. This detailed evaluation is only required if repeated observations indicate gap is actively growing.

Determined by FMEA

Approach:

Generic efforts to support inspection.

- Extension of MRP model to look at relationship between swelling and deformation at seam.
- Guideline for issues to be addressed in plant-specific FMEA.

CE-ID: 6 Core Support Barrel Assembly

Upper (Core Support Barrel) Flange Weld

Category:	Primary Applicability: All plants
Degradation Effect:	Cracking (SCC)
Expansion Link:	Remaining core barrel assembly welds, core support column welds
Function:	Primary core support structure.

Inspection

Method:	Enhanced visual (EVT-1) examination no later than 2 refueling outages from the beginning of the license renewal period. Subsequent examinations on a 10-year interval.
Coverage:	100% of the accessible surfaces of the upper flange weld. See MRP-227 Figure 4-15.
Observable Effect:	Stress corrosion cracking

Failure

Failure Mechanism:	SCC
Failure Effect:	Loss of core support
Failure Criteria:	Actively growing through-wall flaws require mitigation. Require demonstration that flaw growth is arrested or limited to surface. An existing through-wall flaw may be acceptable if condition and shape indicate that it is non-growing fabrication flaw.

Methodology

Goal: Due to the high fracture toughness of unirradiated stainless steel, the core barrel is a highly flaw tolerant structure and flaw sizes are expected to be very large. However, the core barrel is a critical support structure. Flaw growth in this component is outside the range of normal expectations. Therefore, it has been assumed that the presence of any actively growing through-wall crack would require repair or other mitigation. The goal of the calculation is to demonstrate the crack is stable or not likely to grow through wall.

Data Requirements:

1. Operating loads
2. K Solutions for range of expected crack shapes (lengths and depths)
3. SCC crack growth rate curves
4. Fatigue crack growth rate curve (as backup)

Analysis: Strategy similar to Westinghouse core barrel upper flange weld.

Option 1. Observation on OD of core support barrel

Step1. Determine stress distribution through core support barrel thickness for normal operating conditions (expect peak stress at vessel OD).

Step 2. Obtain stress intensity factor solution for part-through-wall crack as function of surface length (L) and depth (a).

Step 3. Short cracks will be constrained by the stress distribution in the barrel wall. Define the maximum constrained crack length as Lc.

CE-ID: 6 **Core Support Barrel Assembly**

Upper (Core Support Barrel) Flange Weld

Step 4. OD crack observation is acceptable if L<Lc.

Step 5. If L > Lc, then must perform UT to determine crack depth (a).

Step 6. Crack is acceptable if K corresponding to a and Lc is less than 20 ksi-in^1/2.

Step 7. All remaining cracks require specific flaw-tolerance analysis.

Option 2. Observation of flaw on ID of core support barrel

Step 1. If flaw on ID is smaller than the length (Lc) defined in Option 1, visually examine the OD surface corresponding to the ID flaw to determine if it is OD-initiated. Crack is acceptable if not through-wall.

Step 2. For a through-wall flaw, apply the OD flaw acceptance criteria from Option 1.

Step 3. All remaining cracks require a geometry-specific flaw-tolerance analysis.

Option 3. Observation of crack on ID of core support barrel

Step 1. If flaw on ID is smaller than the length (Lc) defined in Option 1, perform UT exam to determine if the crack is through-wall. Crack is acceptable if not through-wall.

Step 2. For a through-wall flaw, apply the OD flaw acceptance criteria from Option 1.

Step 3. All remaining cracks require a geometry-specific flaw-tolerance analysis.

Acceptance Criteria: Demonstrate that crack is not actively growing or limited to surface as indicted by analysis.

Approach: Plant-specific analysis

CE-ID: 6.1	**Core Support Barrel Assembly**		
	Lower Core Barrel Flange		
Category:	Expansion	Applicability:	All plants
Degradation Effect:	Cracking (SCC, Fatigue)		
Expansion Link:	Upper (core support barrel) flange weld		
Function:	Primary core support.		

Inspection

Method: Enhanced visual (EVT-1) examination, with initial and subsequent examinations dependent on the results of the upper (core support barrel) flange weld examinations.

Coverage: 100% of accessible welds and adjacent base metal.

See MRP-227 Figure 4-15.

Observable Effect: Cracks

Failure

Failure Mechanism: SCC or fatigue cracking in weld or weld heat affected zone.

Failure Effect: Loss of core support

Failure Criteria: Potential for growth of a through-wall flaw before next planned inspection.

An existing through flaw may be acceptable if condition and shape indicate that it is a non-growing fabrication flaw.

Methodology

Goal: Due to the high fracture toughness of unirradiated stainless steel, the core barrel is a highly flaw tolerant structure and flaw sizes are expected to be very large. However, the core barrel is a critical support structure. Flaw growth in this component is outside the range of normal expectations. Therefore, it has been assumed that the presence of any actively growing through-wall crack would require repair or other mitigation. The goal of the calculation is to demonstrate the crack is stable or not likely to grow through wall.

Data Requirements:

1. Operating loads
2. K Solutions for range of expected crack shapes (lengths and depths)
3. SCC crack growth rate curves
4. Fatigue crack growth rate curve (as backup)

Analysis:

1. Perform FEA to determine stress distribution across weld.
2. Evaluate stress distribution to determine surface with highest probability of crack initiation (highest tensile stress).

CE-ID: 6.1 **Core Support Barrel Assembly**

Lower Core Barrel Flange

3. Establish criteria for most likely surface.

   > K solution for observed crack length indicates diminishing stress intensity with increasing crack length.

   \- or -

   > UT examination indicates that flaw is limited to initiating surface.

4. Establish criteria for least likely surface.

   > Require demonstration that observed flaw was not initiated on opposite surface and grown through wall.

   - No visual evidence of cracking on opposite surface.
   - Visual observation of opposite surface indicates deep, narrow crack inconsistent with actively growing mechanism.
   - UT exam indicates that flaw is limited to initiating surface.

5. Flaws that do not meet criteria of 3 and 4 require additional geometry-specific analysis to estimate rate of crack growth and establish acceptable crack lengths. Reduced inspection intervals may be required.

Acceptance Criteria: Current crack size is explainable by known crack growth rate laws and limited crack growth is projected.

Approach: Plant-specific analysis.

- Require flaw tolerance handbook/methodology based on flaw location and direction.
- MRP-210 may have limited relevance.

CE-ID: 6.2 Core Support Barrel Assembly

Remaining Core Barrel Assembly Welds

Category:	Expansion	Applicability:	Core support barrel assembly
Degradation Effect:	Cracking (SCC)		
Expansion Link:	Upper (core support barrel) flange weld		
Function:	Primary core support		

Inspection

Method: Enhanced visual (EVT-1) examination, with initial and subsequent examinations dependent on the results of core barrel assembly upper flange weld examinations.

Coverage: 100% of one side of the accessible weld and adjacent base metal surfaces for the weld with the highest calculated operating stress.

Observable Effect: Cracks

Failure

Failure Mechanism: Due to the high fracture toughness of unirradiated stainless steel, the core barrel is a highly flaw tolerant structure and critical flaw sizes are expected to be very large. However, the core barrel is a critical core support structure. Flaw initiation and growth in this component is outside the range of normal expectations. Therefore, it has been assumed that the presence of any actively growing through-wall flaw would require repair or other mitigation.

Failure Effect: Potential loss of core support

Failure Criteria: Potential for growth of a through-wall flaw before next planned inspection. An existing through flaw may be acceptable if condition and shape indicate that it is a non-growing fabrication flaw.

Methodology

Goal: Demonstrate that observed flaws are not actively growing.

Data Requirements:

1. Fast neutron fluence (dpa)
2. Irradiated fracture toughness
3. Operating loads
4. K Solutions for range of expected crack shapes (lengths and depths)
5. SCC crack growth rate curves
6. Fatigue crack growth rate curve (as backup)

CE-ID: 6.2 Core Support Barrel Assembly

Remaining Core Barrel Assembly Welds

Analysis:

1. Flaws in core support barrel above the shroud section will be evaluated assuming active crack growth mechanisms are SCC and fatigue.
2. Flaws in the beltline region of the core support barrel (shroud section) will be evaluated assuming active growth mechanisms are IASCC and fatigue.
3. A fluence estimate at the flaw location is required for all flaws in the beltline region.
4. Normal operating and fatigue loads will be established for core barrel at this location.
5. Determine stress intensity factors for a through-wall crack.
6. Use appropriate crack growth rate models (SCC or IASCC and fatigue) to estimate crack growth rate.
7. If crack growth rate is consistent with observed flaw size:
   - Project flaw size through inspection interval using crack growth rate estimate.
   - Determine loads during limiting transient.
   - Determine stress intensity factor for through-wall crack of projected length.
   - For low fluence region assume K_{Ic} = 150 ksi-in^1/2
   - For beltline region determine lower bound toughness based on fluence estimate.
   - If stress intensity factor during transient is less than fracture toughness, flaw is acceptable.
   - If stress intensity factor during transient is greater than fracture toughness, proceed to Step 8.
8. If crack growth rate is too low to explain existence of observed crack or flaw not acceptable by Step 7:
   - Determine crack depth.
   - If crack depth small compared to barrel thickness (<xx inches), then crack is acceptable.
   - If crack depth large compared to barrel thickness, the crack is rapidly growing and a detailed analysis is required.

Acceptance Criteria: Current crack size is explainable by known crack growth rate laws and limited crack growth is projected.

Approach: Plant-specific analysis. (See item CE-ID 6.1)

- Require flaw tolerance handbook/methodology based on flaw location and direction.
- MRP-210 may have limited relevance.

CE-ID: 6.3	Lower Support Structure
	Core Support Column Welds

Category: Expansion — Applicability: *All plants except those with core shrouds assembled with full-height shroud*

Degradation Effect: Cracking (SCC, IASCC, fatigue) including damaged or fractured material

Expansion Link: Upper (core support barrel) flange weld

Function: The support columns are a primary core support structure. The columns keep the core support plate from sagging or excess thermal deformation.

Inspection

Method: Visual (VT-3) examination, with initial and subsequent examinations based on plant evaluation of SCC susceptibility and demonstration of remaining fatigue life.

Coverage: Examination coverage determined by plant-specific analysis.

See MRP-227 Figures 4-16 and 4-31.

Observable Effect: Fracture

Potential for fuel assembly misalignment

Failure

Failure Mechanism: Cracking

Failure Effect: Failure of support columns will allow local deformation of core support plate.

Cracks initiating in welds may lead to fracture or loss of attachment to core support plate.

Failure Criteria: Must establish minimum core support column distributions required to maintain core support plate stability. (Alternative would be to demonstrate that a limited number (5) of failures are generally acceptable.)

Methodology

Goal: Establish minimum acceptable pattern of core support columns.

Data Requirements:

1. Design criteria used to determine number and spacing of core supports: lower core support plate loads during normal operating and limiting transient conditions, etc.
2. Loads on lower core plate
3. Fluence accumulated by the core support columns
4. Constitutive model for stainless steel properties as a function of irradiation
5. Displacement tolerances on core support plate
6. Geometry

Analysis:

1. Establish functional requirements for core support columns.
   A. During normal operation, the system of support columns should resist core plate deformation due to mechanical or thermal loading. Core plate requirements for "flatness" and fuel assembly alignment.
   B. During limiting accident transient, the system must maintain structural integrity.

CE-ID: 6.3 **Lower Support Structure**

Core Support Column Welds

2. Support column analysis assumptions.

   A. Assume any column with a crack in main body to have failed.

   B. Assume any column with a crack in the weld to result in failure of the attachment.

3. Structural model of lower support structure.

   Model of lower support structure that includes support columns and lower core plate. Model should be capable of removing individual column or breaking attachment to lower core plate. Would require multiple iterations to establish "minimum acceptable patterns" of core support columns and support column welds.

4. Structural model must be run for functional requirements A and B.

5. Determine margin for additional failures.

Assume number of failures in next 10 years is equal to number observed to date.

N = # of Support Columns

Nf = # of Observed Flawed Columns

Nreq = # of columns in relevant minimum pattern

$$\text{Margin} = N - N_{req}$$

Acceptance Criteria: Require that no more of 1/2 of columns in margin are failed:

$$N_f < (N - N_{req})/2$$

Approach: Generic program to share first-of-a-kind effort. (See W-ID: 2.1 and CE-ID: 1.2)

- Pilot analysis of lower support structure to identify critical issues.
- Expect final acceptance based on plant-specific analysis.

CE-ID: 7	**Core Support Barrel Assembly**		
	Lower Flange Weld		
Category:	Primary	Applicability:	All plants
Degradation Effect:	Cracking (fatigue)		
Expansion Link:	None		
Function:	Primary core support structure		

Inspection

Method:	If fatigue life cannot be demonstrated by time-limited aging analysis (TLAA), enhanced visual (EVT-1) examination is required no later than 2 refueling outages from the beginning of the license renewal period. Subsequent examination on a 10-year interval.
Coverage:	Examination coverage to be defined by plant-specific fatigue analysis. See MRP-227 Figure 4-15.
Observable Effect:	Cracking

Failure

Failure Mechanism:	Fatigue
Failure Effect:	Loss of core support
Failure Criteria:	Potential for growth of a through-wall flaw before next planned inspection. An existing through-wall flaw may be acceptable if condition and shape indicate that it is a non-growing fabrication flaw.

Methodology

Goal: Demonstrate that observed flaws are not actively growing.

Data Requirements:

1. Operating loads
2. K solutions for range of expected crack shapes (lengths and depths)
3. Fatigue crack growth rate curves
4. SCC crack growth rate curve (as backup)

Analysis:

Inspection of this item is required if sufficient fatigue life cannot be demonstrated by normal time-limited aging analysis (TLAA) procedures. Due to general concerns about SCC in structural welds, the same location has been listed as an expansion inspection that would be triggered by observation of cracking in the upper flange weld.

A general outline of TLAA procedures is provided separately. The TLAA process evaluates potential fatigue crack initiation. As part of that evaluation the stress amplitude and frequency must be estimated. If the TLAA indicates that crack initiation is possible, inspection of the indicated locations is required. Fatigue crack growth rates used in establishing acceptance criteria for the inspections should be based on the stress amplitudes and frequencies used in the TLAA.

CE-ID: 7 **Core Support Barrel Assembly**

Lower Flange Weld

The following analysis parallels the requirements for the expansion inspections.

1. Perform FEA to determine stress distribution across weld.
2. Evaluate stress distribution to determine surface with highest probability of crack initiation (highest tensile stress).
3. Establish criteria for the highest probability surface.
   - Demonstrate that a 1/4 thickness flaw of observed length will not grow through barrel wall in planned inspection interval.
4. Establish criteria for the lowest probability surface.
   - Require demonstration that observed flaw was not initiated on opposite surface and grown through wall.
     - No visual evidence of cracking on opposite surface.
     - Visual observation of opposite surface indicates deep, narrow crack inconsistent with actively growing mechanism.
     - UT exam indicates that flaw is limited to initiating surface.
5. Flaws that do not meet criteria of Items 3 and 4 require additional geometry-specific analysis to estimate rate of crack growth and establish acceptable crack lengths. Reduced inspection intervals may be required.

Acceptance Criteria: Acceptance criteria for TLAA related items are beyond scope of current project.

Approach: TLAA (plant specific)

Potential flaw analysis if inspection required.

- Require flaw tolerance handbook/methodology based on flaw location and direction.
- MRP-210 may have limited relevance.

CE-ID: 8	**Lower Support Structure**		
	Core Support Plate		
Category:	Primary	Applicability:	All plants with a core support plate
Degradation Effect:	Cracking (fatigue)		
Expansion Link:	None		
Function:	Primary core support. Provides alignment of fuel assembly.		

Inspection

Method:	If fatigue life cannot be demonstrated by time-limited aging analysis (TLAA), enhanced visual (EVT-1) examination is required no later than 2 refueling outages from the beginning of the license renewal period. Subsequent examination on a 10-year interval.
Coverage:	Examination coverage to be defined by plant-specific fatigue analysis. See MRP-227 Figure 4-16.
Observable Effect:	Cracking

Failure

Failure Mechanism:	Fatigue
Failure Effect:	1. Loss of core support 2. Difficulty in loading fuel due to misalignment
Failure Criteria:	Displacement of core support plate

Methodology

Goal:	Cracks in the core support plate are expected to grow from hole-to-hole within the plate. A network of connected cracks is required to allow significant displacement in the plate. The goal is to demonstrate that the cracking present does not cause enough displacement to affect fuel loading or core support.
Data Requirements:	1. Operating loads 2. K Solutions for range of expected crack shapes (lengths and depths) 3. Fatigue crack growth rate curves 4. IASCC crack growth rate curve and fluence (as backup)
Analysis:	Inspection of this item is only required if sufficient fatigue life cannot be demonstrated by normal time-limited aging analysis (TLAA) procedures. A general outline of TLAA procedures is provided separately. The TLAA process evaluates potential fatigue crack initiation. As part of that evaluation the stress amplitude and frequency must be estimated. If the TLAA indicates that crack initiation is possible, inspection of the indicated locations is required. Fatigue crack growth rates used in establishing acceptance criteria for the inspections should be based on the stress amplitudes and frequencies used in the TLAA.

CE-ID: 8

Lower Support Structure

Core Support Plate

Process steps for establishing allowable crack length in the core support plate:

1. Establish functional requirements for core support plate.

   A. During normal operation, the system of support columns should resist core support plate deformation caused by mechanical or thermal loading. The core support plate would have requirements for "flatness" and fuel assembly alignment.

   B. During limiting accident transient system must maintain structural integrity.

2. Core support plate analysis assumptions.

   A. Assume crack initiates at the hole or holes in plate with highest surface tensile stress.

   B. Assume crack propagates to the adjacent hole with highest stress.

3. Structural model of lower support structure.

   Model of lower support structure that includes support columns and lower core support plate. Model should be capable of modeling a crack connecting holes in plate (crack tip modeling not required). Evaluate displacement on the surface of the core support plate.

4. Any single observed crack is acceptable if displacement across crack in FEA model meets design requirements for plate.

5. If unable to demonstrate acceptability of a single crack, require detailed flaw analysis.

6. Optional determination of margin for additional cracking. Repeat evaluation for multiple cracks connecting adjacent holes. Determine number and pattern of connected holes to violate design requirements.

Acceptance Criteria: Acceptance criteria for TLAA related items are beyond the scope of current project.

Approach: TLAA (plant specific)

CE-ID: 9	**Upper Internals Assembly** **Fuel Alignment Plate**		
Category:	Primary	Applicability:	All plants with core shrouds assembled with full-height shroud plates
Degradation Effect:	Cracking (fatigue)		
Expansion Link:	None		
Function:	Provide fuel assembly alignment and support. Direct flow into upper internals.		

Inspection

Method:	If fatigue life cannot be demonstrated by time-limited aging analysis (TLAA), enhanced visual (EVT-1) examination is required no later than 2 refueling outages from the beginning of the license renewal period. Subsequent examination on a 10-year interval.
Coverage:	Examination coverage to be defined by plant-specific fatigue analysis. See MRP-227 Figure 4-17.
Observable Effect:	TLAA should be completed prior to inspection program. Normal rules for demonstrating fatigue life should be applied with updated projections of the number of load cycles. Visual inspections for fatigue cracks along weld are required if sufficient fatigue life can not be demonstrated

Failure

Failure Mechanism:	1. Linkage of cracks from multiple origination sites leads to loss of integrity in fuel alignment plate. 2. Crack displacement causes misalignment of fuel assemblies.
Failure Effect	1. Loss of structural stability 2. Difficulty in loading fuel due to misalignment
Failure Criteria:	1. Linkage of cracks that will create a critical flaw length 2. Linkage of cracks that will allow vertical displacement of a section of the fuel alignment plate

Methodology

Goal:	Demonstrate that cracking of fuel alignment plate will not cause significant problems loading fuel.
Data Requirements:	Stress analysis results for fatigue loading.
Analysis:	Inspection of this item is required if sufficient fatigue life cannot be demonstrated by normal time-limited aging analysis (TLAA) procedures. A general outline of TLAA procedures is provided separately. The TLAA process evaluates potential fatigue crack initiation. As part of that evaluation, the stress amplitude and frequency must be estimated. If the TLAA indicates that crack initiation is possible, inspection of the indicated locations is required. Fatigue crack growth rates used in establishing acceptance criteria for the inspections should be based on the stress amplitudes and frequencies used in the TLAA.
Acceptance Criteria:	Acceptance criteria for TLAA related items are beyond scope of current project.
Approach:	TLAA (plant specific – applies to one utility)

CE-ID: 10	**Control Element Assembly Instrument Guide Tubes**		
Category:	Primary	Applicability:	All plants with instrument guide tubes in the CEA shroud assembly
Degradation Effect:	Cracking (SCC, fatigue) that results in missing supports or separation at the welded joint		
Expansion Link:	Remaining instrument guide tubes within the CEA shroud assemblies		
Function:	Define path for insertion of in-core instrumentation.		

Inspection

Method:	Visual (VT-3) examination, no later than 2 refueling outages from the beginning of the license renewal period. Subsequent examination on a 10-year interval. Plant-specific component integrity assessments may be required if degradation is detected and remedial action is needed.
Coverage:	100% of tubes in peripheral CEA shroud assemblies (i.e., those adjacent to the perimeter of the fuel alignment plate). See MRP-227 Figure 4-18.
Observable Effect:	Missing or broken supports.

Failure

Failure Mechanism:	Cracking
Failure Effect:	1. Potential loose parts 2. Inability to insert/withdraw instrumentation
Failure Criteria:	1. Potential uncontained loose parts 2. Inability to maintain minimum in-core instrumentation

Methodology

Goal:	Demonstrate ability to insert instrumentation.
Data Requirements:	Instrumentation requirements for plant.
Analysis:	1. Evaluate stability of failed instrument guide tube. Any section that could potentially detach and become a loose part or otherwise interfere with plant operation should be removed or stabilized. 2. Any instrument guide tube with an observable crack will be assumed to have failed.
Acceptance Criteria:	1. Configuration of unfailed guide tubes should be sufficient to allow adequate core monitoring. 2. No margin is required for this item. If the instrumentation is functional at start-up, the plant can be operated.
Approach:	Pass/Fail inspection with established minimum number of instrumentation tubes. Based directly on plant specifications.

CE-ID: 10.1	**Control Element Assembly**		
	Remaining Instrument Guide Tubes		
Category:	Expansion	Applicability:	All plants with instrument guide tubes in the CEA shroud assembly
Degradation Effect:	Cracking		
Expansion Link:	Peripheral instrument guide tubes within the CEA shroud assemblies		
Function:	Define path for insertion of in-core instrumentation.		

Inspection

Method: Visual (VT-3) examination, with initial and subsequent examinations dependent on the results of the instrument guide tubes examinations.

Coverage: 100% of tubes in CEA shroud assemblies.

See MRP-227 Figure 4-18.

Observable Effect: Missing or broken supports

Failure

Failure Mechanism: Cracking of attachment welds

Failure Effect:
1. Potential loose parts
2. Inability to insert/withdraw instrumentation

Failure Criteria:
1. Potential uncontained loose parts
2. Inability to maintain minimum in-core instrumentation

Methodology

Goal: Demonstrate ability to insert instrumentation.

Data Requirements: Instrumentation requirements for plant.

Analysis:
1. Evaluate stability of failed instrument guide tube. Any section that could potentially detach and become a loose part or otherwise interfere with plant operation should be removed or stabilized.
2. Any instrument guide tube with an observable crack will be assumed to have failed.

Acceptance Criteria:
1. Configuration of unfailed guide tubes should be sufficient to allow adequate core monitoring.
2. No margin is required for this item. If the instrumentation is functional at start-up, the plant can be operated.

Approach: Pass/Fail inspection with established minimum number of instrumentation tubes. Based directly on plant specifications. (See CE-ID: 10)

CE-ID: 11	**Lower Support Structure**		
	Deep Beams		
Category:	Primary	Applicability:	All plants with core shrouds assembled with full-height shroud plates
Degradation Effect:	Cracking (fatigue) Check for a detectable surface-breaking indication in the welds		
Expansion Link:	None		
Function:	Support core. Direct coolant flow into core.		

Inspection

Method:	Enhanced visual (EVT-1) examination, no later than 2 refueling outages from the beginning of the license renewal period. Subsequent examination on a 10-year interval, if adequacy of remaining fatigue life cannot be demonstrated by TLAA.
Coverage:	Examine beam-to-beam welds in the axial elevation from the beam top surface to four inches below. See MRP-227 Figure 4-19.
Observable Effect:	Fatigue crack growth along welds at beams. Check for a detectable surface-breaking indication in the welds or beams.

Failure

Failure Mechanism:	Cracking
Failure Effect:	Loss of fuel assembly alignment
Failure Criteria:	No cracking that will cause displacement of fuel alignment pins

Methodology

Goal:	Demonstrate stability of lower support structure.
Data Requirements:	Potential fatigue loading and cycles.
Analysis:	Inspection of this item is required if sufficient fatigue life cannot be demonstrated by normal time-limited aging analysis (TLAA) procedures. A general outline of TLAA procedures is provided separately. The TLAA process evaluates potential fatigue crack initiation. As part of that evaluation, the stress amplitude and frequency must be estimated. If the TLAA indicates that crack initiation is possible, inspection of the indicated locations is required. Fatigue crack growth rates used in establishing acceptance criteria for the inspections should be based on the stress amplitudes and frequencies used in the TLAA.

CE-ID: 11

Lower Support Structure

Deep Beams

The general analysis of this structure would address the following issues:

1. The grid structure of the lower core support in these plants precludes catastrophic failure initiated by a single crack.
2. Cracking that will not result in failure of any beam or structural weld within the planned inspection interval should be acceptable.
   - Assume crack initiation at most probable location as defined by TLAA.
   - Evaluate crack depth (a)
     - Determine crack growth rate consistent with stress amplitude and frequency used in TLAA.
     - Project crack growth through planned inspection interval.
   - Plot projected remaining ligament as a function of crack depth.
   - Maximum acceptable crack size corresponds to projected remaining ligament = 0.
3. Additional margin against failure not required (catastrophic failure unlikely).

Acceptance Criteria: Acceptance criteria for TLAA related items are beyond scope of current project.

Approach: TLAA (plant specific – applies to one utility)

APPENDIX D
FLOW CHARTS OF ILLUSTRATING EVALUATION METHODOLOGIES FOR COMBUSTION ENGINEERING-DESIGNED PLANTS

CE Primary and Expansion Components

CE-ID: 1 Core Shroud Assembly (Bolted) – Core Shroud Bolts

- CE-ID: 1.1 Core Shroud Assembly (Bolted) – Barrel-Shroud Bolts
- CE-ID: 1.2 Core Shroud Assembly (Bolted) – Core Support Column Bolts

CE-ID: 2 Core Shroud Assembly (Welded) – Welds

- CE-ID: 2.1 Core Shroud Assembly (Welded) – Remaining Axial Welds

CE-ID: 3 Core Shroud Assembly (Welded – Full Height) – Shroud Plates

- CE-ID: 3.1 Core Shroud Assembly (Welded – Full Height) – Axial Welds, Ribs and Rings

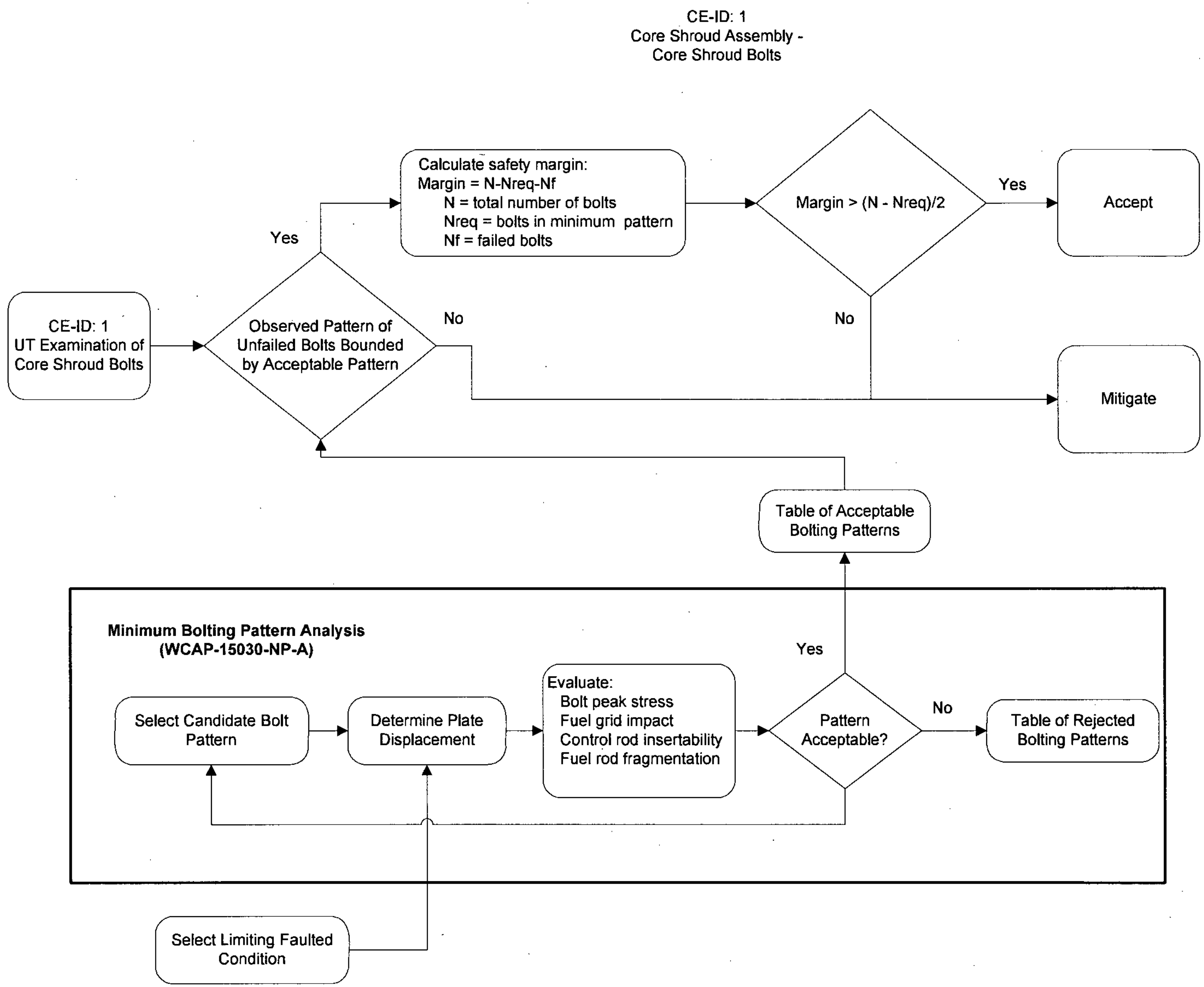
CE-ID: 4 Core Shroud Assembly (Bolted) – Assembly
CE-ID: 5 Core Shroud Assembly (Welded) – Assembly
CE-ID: 6 Core Support Barrel Assembly – Upper (Core Support Barrel) Flange Weld

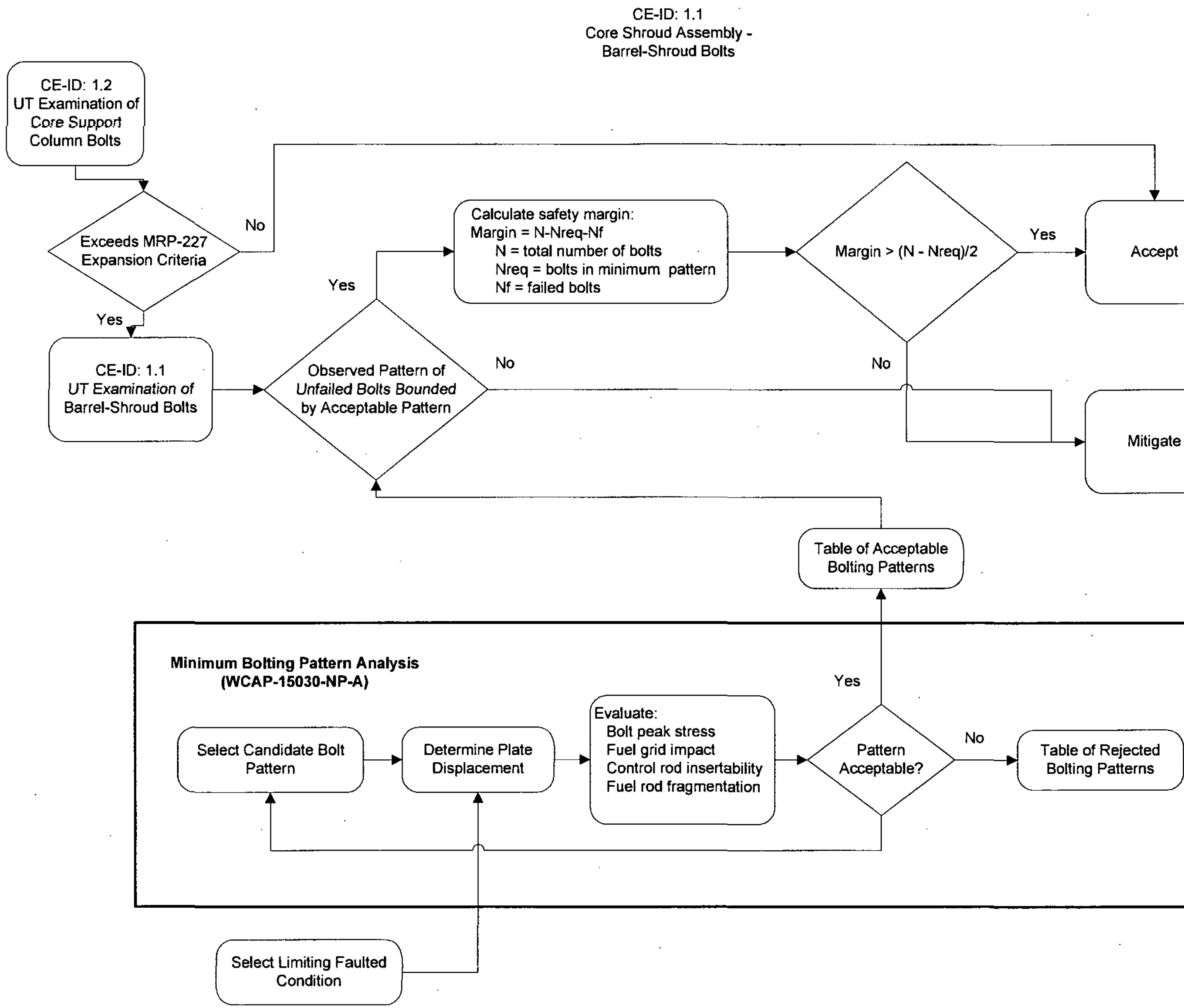
- CE-ID: 6.1 Core Support Barrel Assembly – Lower Core Barrel Flange
- CE-ID: 6.2 Core Support Barrel Assembly – Remaining Core Barrel Assembly Welds
- CE-ID: 6.3 Lower Support Structure – Core Support Column Welds

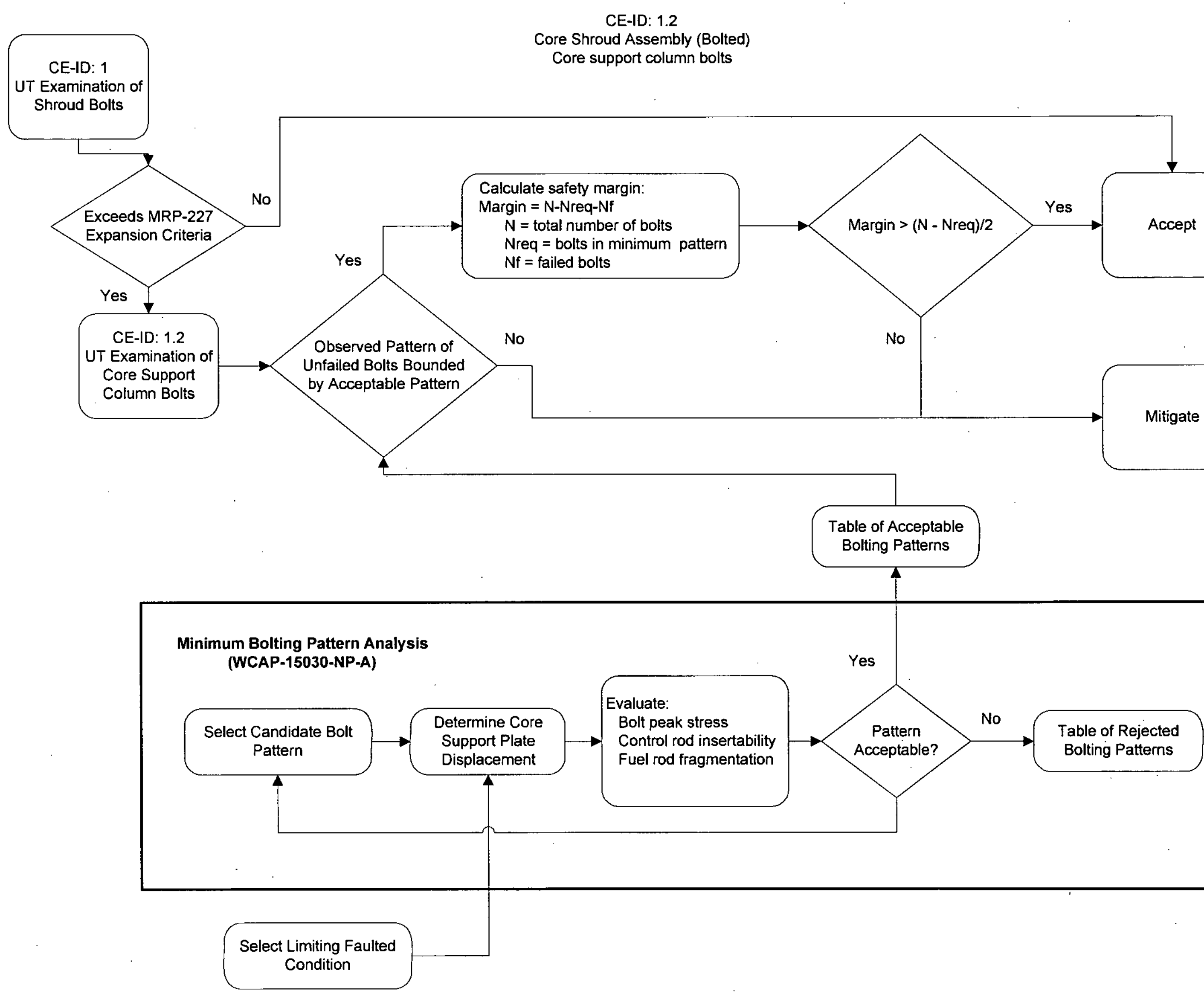
CE-ID: 7 Core Support Barrel Assembly – Lower Flange Weld (Require TLAA – No Figure)
CE-ID: 8 Lower Support Structure – Core Support Plate (Require TLAA – No Figure)
CE-ID: 9 Upper Internals Assembly – Fuel Alignment Plate (Require TLAA – No Figure)
CE-ID: 10 Control Element Assembly – Instrument Guide Tubes

- CE-ID: 10.1 Control Element Assembly – Remaining Instrument Guide Tubes

CE-ID: 11 Lower Support Structure – Deep Beams (Require TLAA – No Figure)


CE-ID: 1
Core Shroud Assembly -
Core Shroud Bolts
CE-ID: 1
UT Examination of
Core Shroud Bolts
Observed Pattern of
Unfailed Bolts Bounded
by Acceptable Pattern
Yes
No
Calculate safety margin:
Margin = N-Nreq-Nf
N = total number of bolts
Nreq = bolts in minimum pattern
Nf = failed bolts
Margin > (N - Nreq)/2
Yes
No
Accept
Mitigate
Table of Acceptable
Bolting Patterns
Minimum Bolting Pattern Analysis
(WCAP-15030-NP-A)
Select Candidate Bolt
Pattern
Determine Plate
Displacement
Evaluate:
Bolt peak stress
Fuel grid impact
Control rod insertability
Fuel rod fragmentation
Pattern
Acceptable?
Yes
No
Table of Rejected
Bolting Patterns
Select Limiting Faulted
Condition

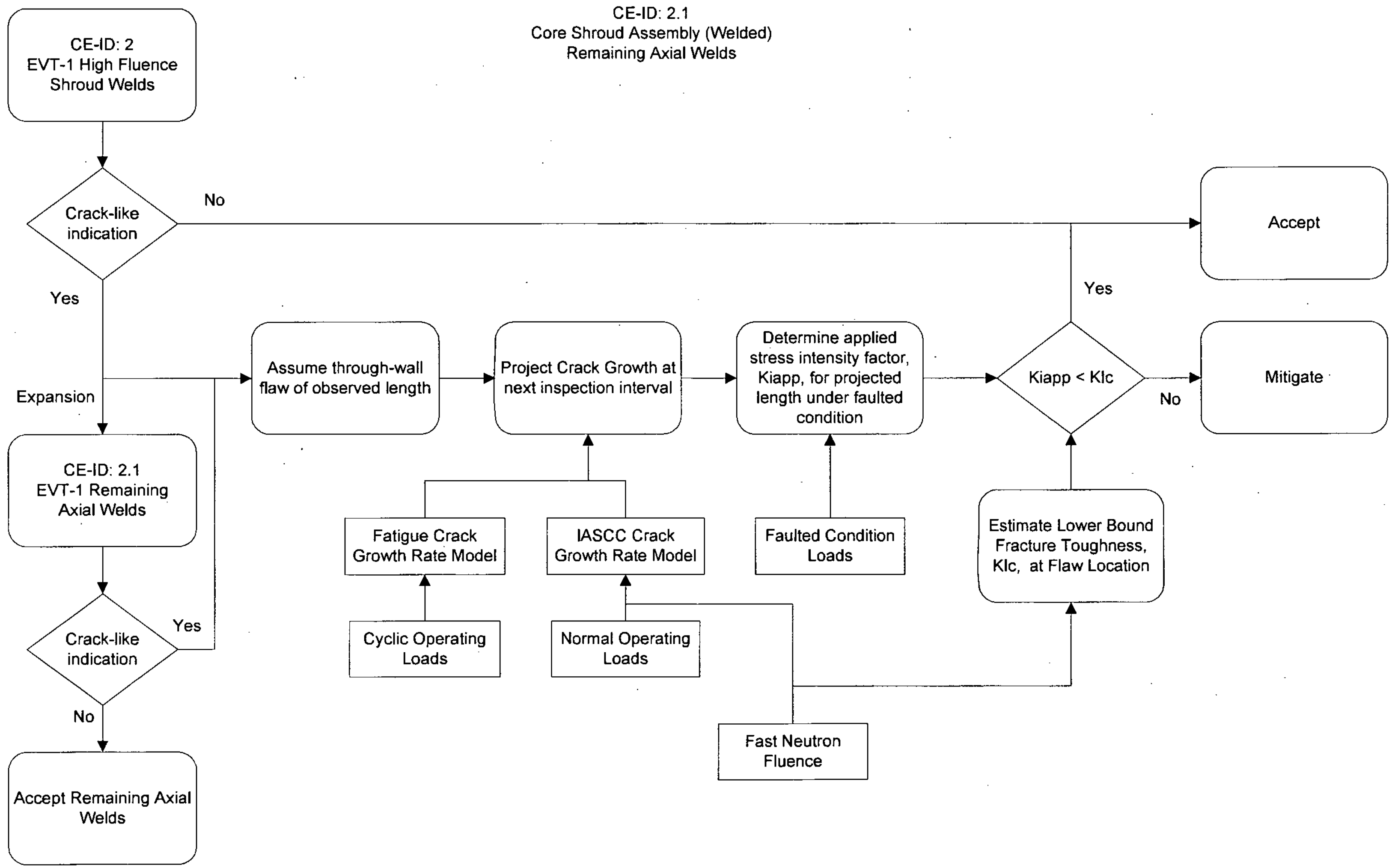

CE-ID: 1.1
Core Shroud Assembly -
Barrel-Shroud Bolts
CE-ID: 1.2
UT Examination of
Core Support
Column Bolts
Exceeds MRP-227
Expansion Criteria
No
Yes
CE-ID: 1.1
UT Examination of
Barrel-Shroud Bolts
Observed Pattern of
Unfailed Bolts Bounded
by Acceptable Pattern
Yes
No
Calculate safety margin:
Margin = N-Nreq-Nf
N = total number of bolts
Nreq = bolts in minimum pattern
Nf = failed bolts
Margin > (N - Nreq)/2
Yes
No
Accept
Mitigate
Table of Acceptable
Bolting Patterns
Minimum Bolting Pattern Analysis
(WCAP-15030-NP-A)
Select Candidate Bolt
Pattern
Determine Plate
Displacement
Evaluate:
Bolt peak stress
Fuel grid impact
Control rod insertability
Fuel rod fragmentation
Pattern
Acceptable?
Yes
No
Table of Rejected
Bolting Patterns
Select Limiting Faulted
Condition


CE-ID: 1.2
Core Shroud Assembly (Bolted)
Core support column bolts
CE-ID: 1
UT Examination of
Shroud Bolts
Exceeds MRP-227
Expansion Criteria
No
Yes
CE-ID: 1.2
UT Examination of
Core Support
Column Bolts
Observed Pattern of
Unfailed Bolts Bounded
by Acceptable Pattern
Yes
No
Calculate safety margin:
Margin = N-Nreq-Nf
N = total number of bolts
Nreq = bolts in minimum pattern
Nf = failed bolts
Margin > (N - Nreq)/2
Yes
No
Accept
Mitigate
Table of Acceptable
Bolting Patterns
Minimum Bolting Pattern Analysis
(WCAP-15030-NP-A)
Select Candidate Bolt
Pattern
Determine Core
Support Plate
Displacement
Evaluate:
Bolt peak stress
Control rod insertability
Fuel rod fragmentation
Pattern
Acceptable?
Yes
No
Table of Rejected
Bolting Patterns
Select Limiting Faulted
Condition

CE-ID: 2
Core Shroud Assembly (Welded)
Shroud Plate

and

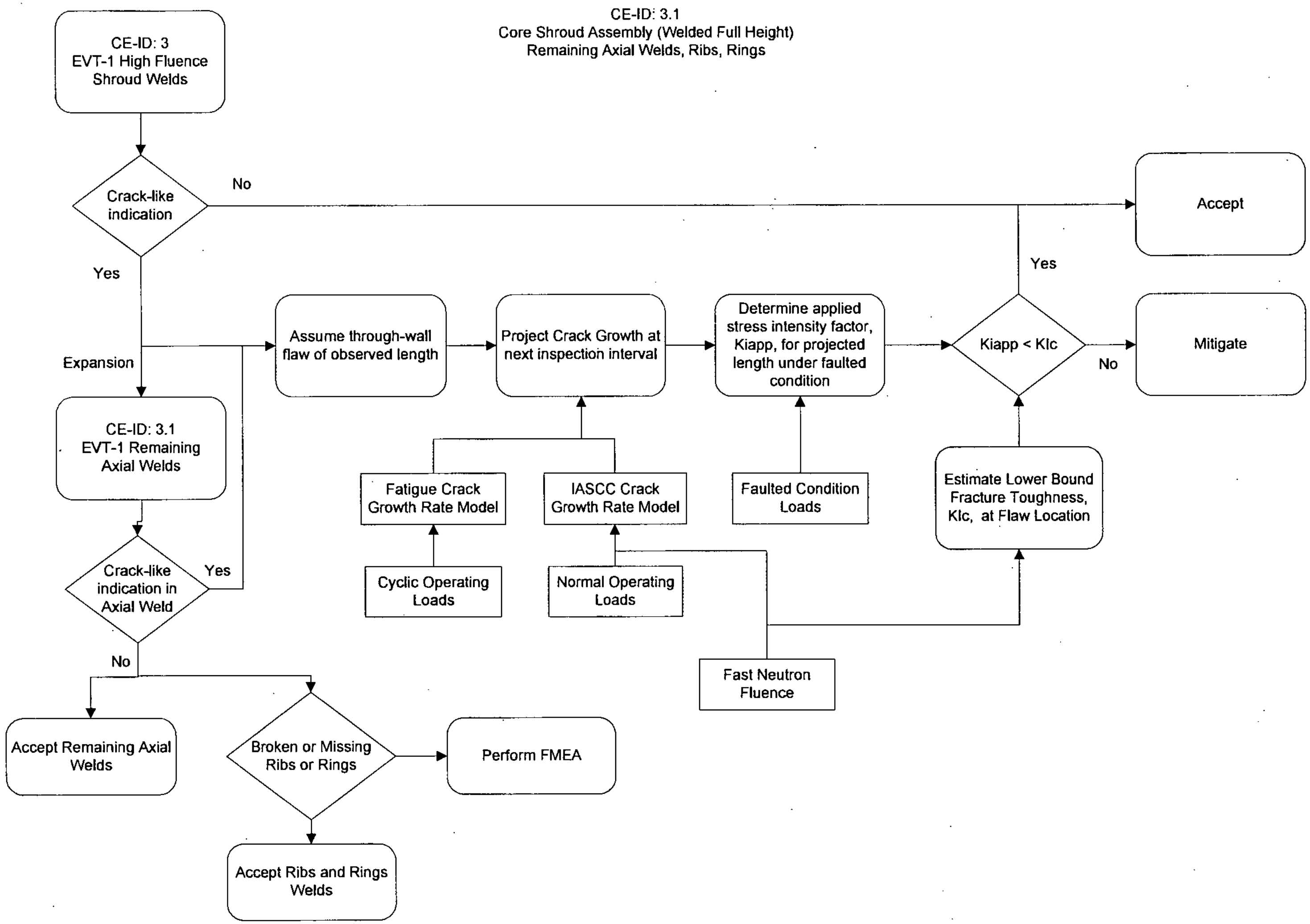
CE-ID: 2.1
Core Shroud Assembly (Welded)
Remaining Axial Welds



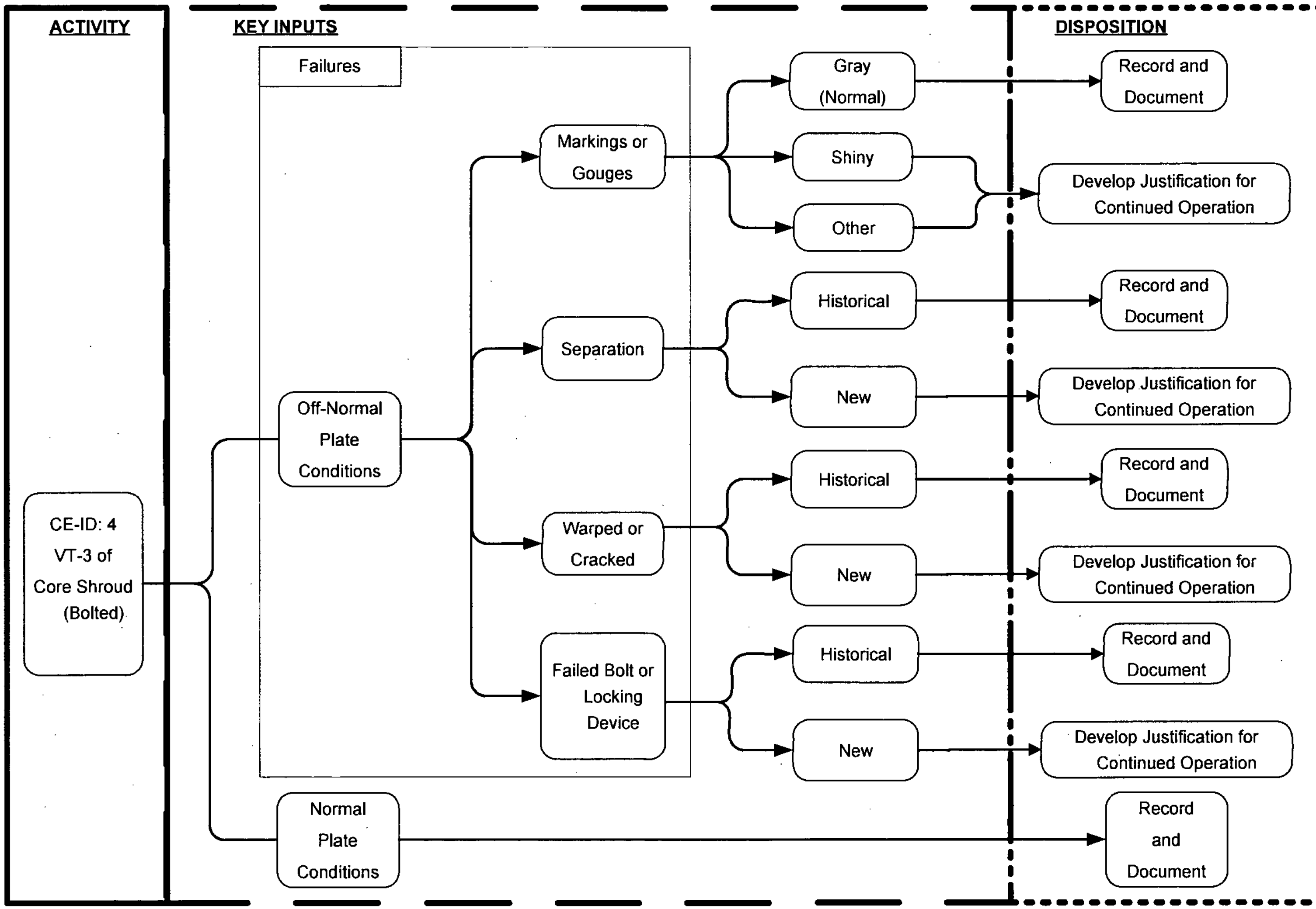
CE-ID: 3
Core Shroud Assembly (Welded-Full Height)
Shroud Plate

and

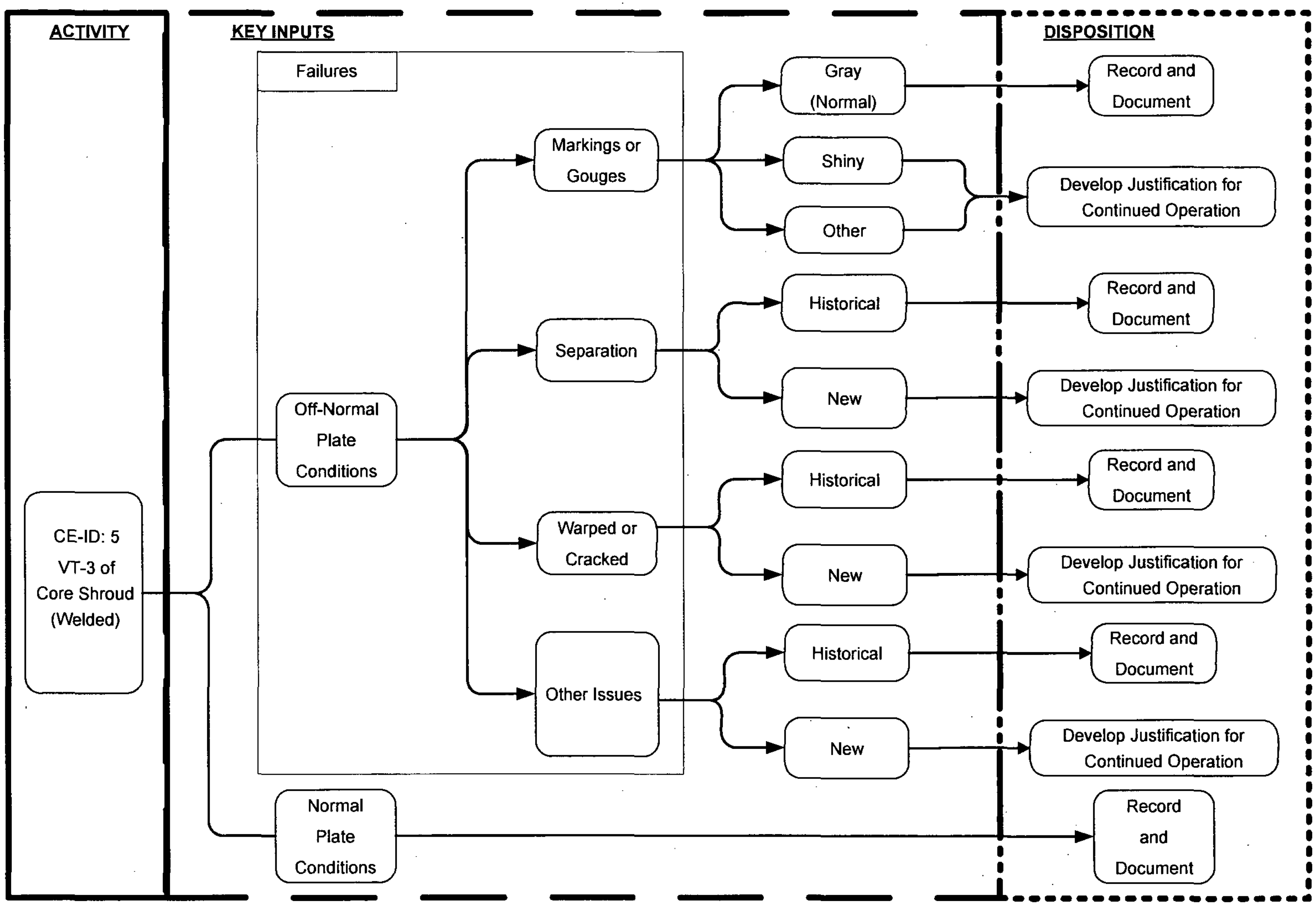
CE-ID: 3.1
Core Shroud Assembly (Welded Full Height)
Remaining Axial Welds, Ribs, Rings



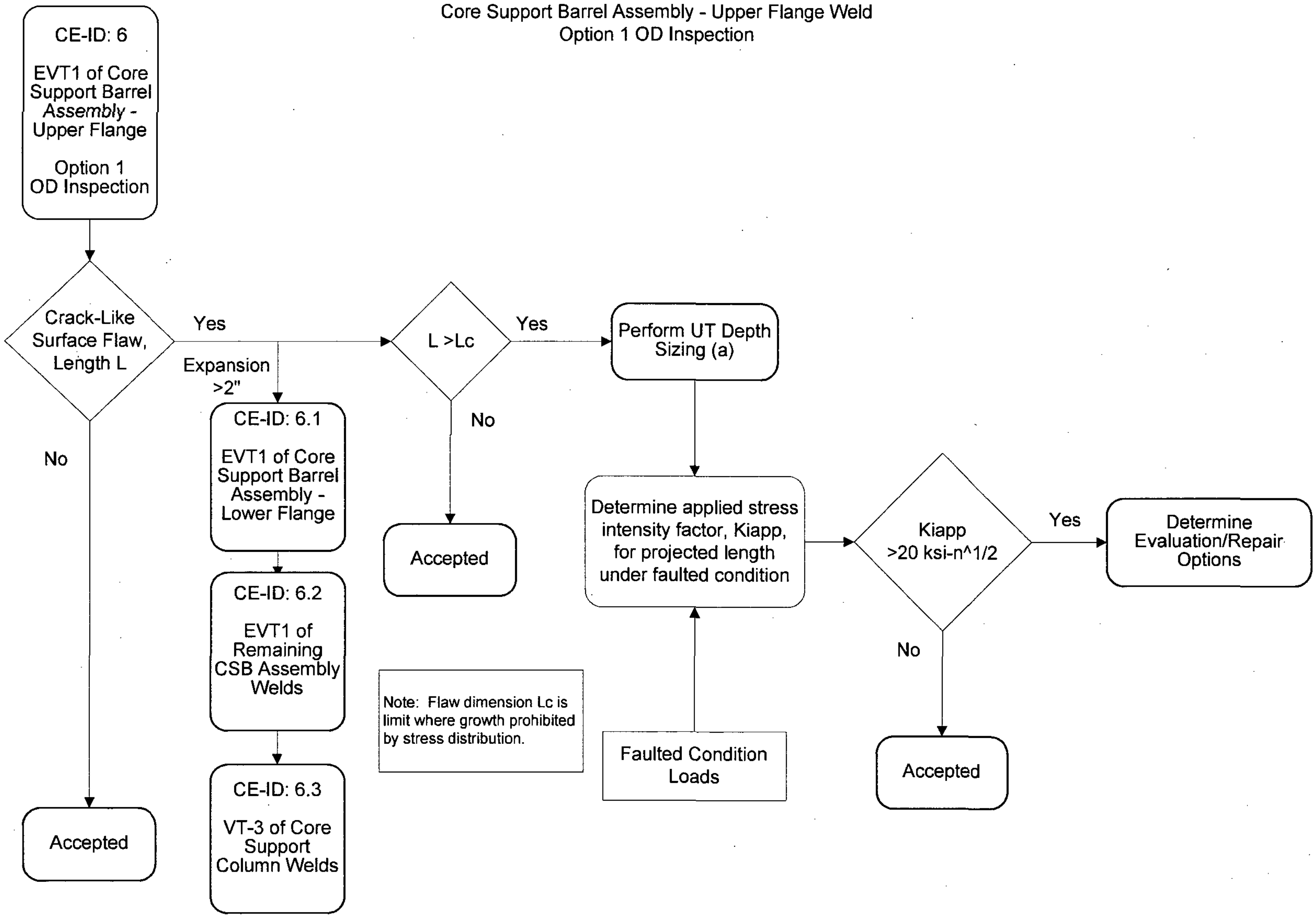
CE-ID:4
Core Shroud Assemby (Bolted)
FMEA Layout



CE-ID:5
Core Shroud Assemby (Welded)
FMEA Layout



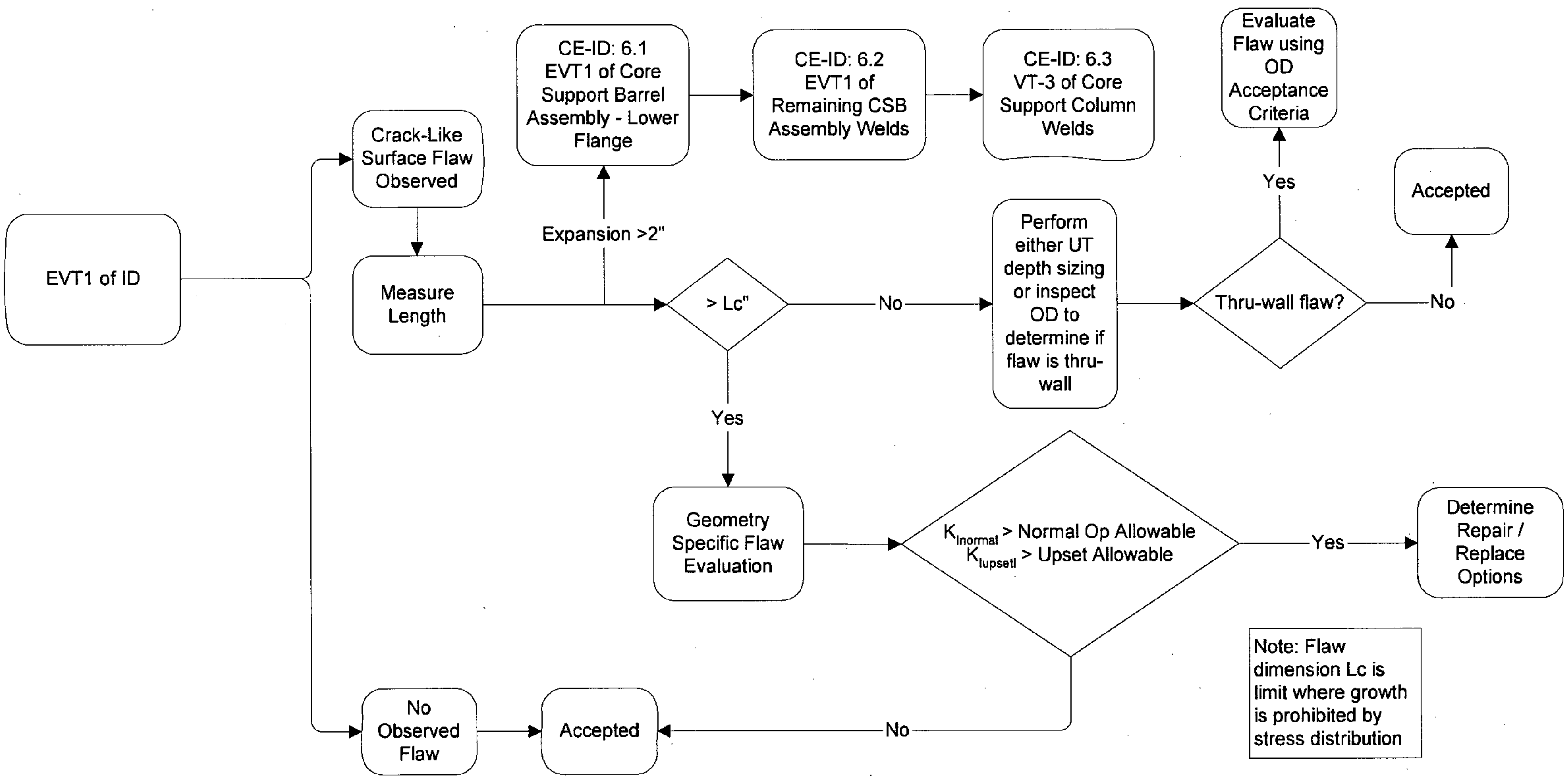
CE-ID:6
Core Support Barrel Assembly - Upper Flange Weld
Option 1 OD Inspection

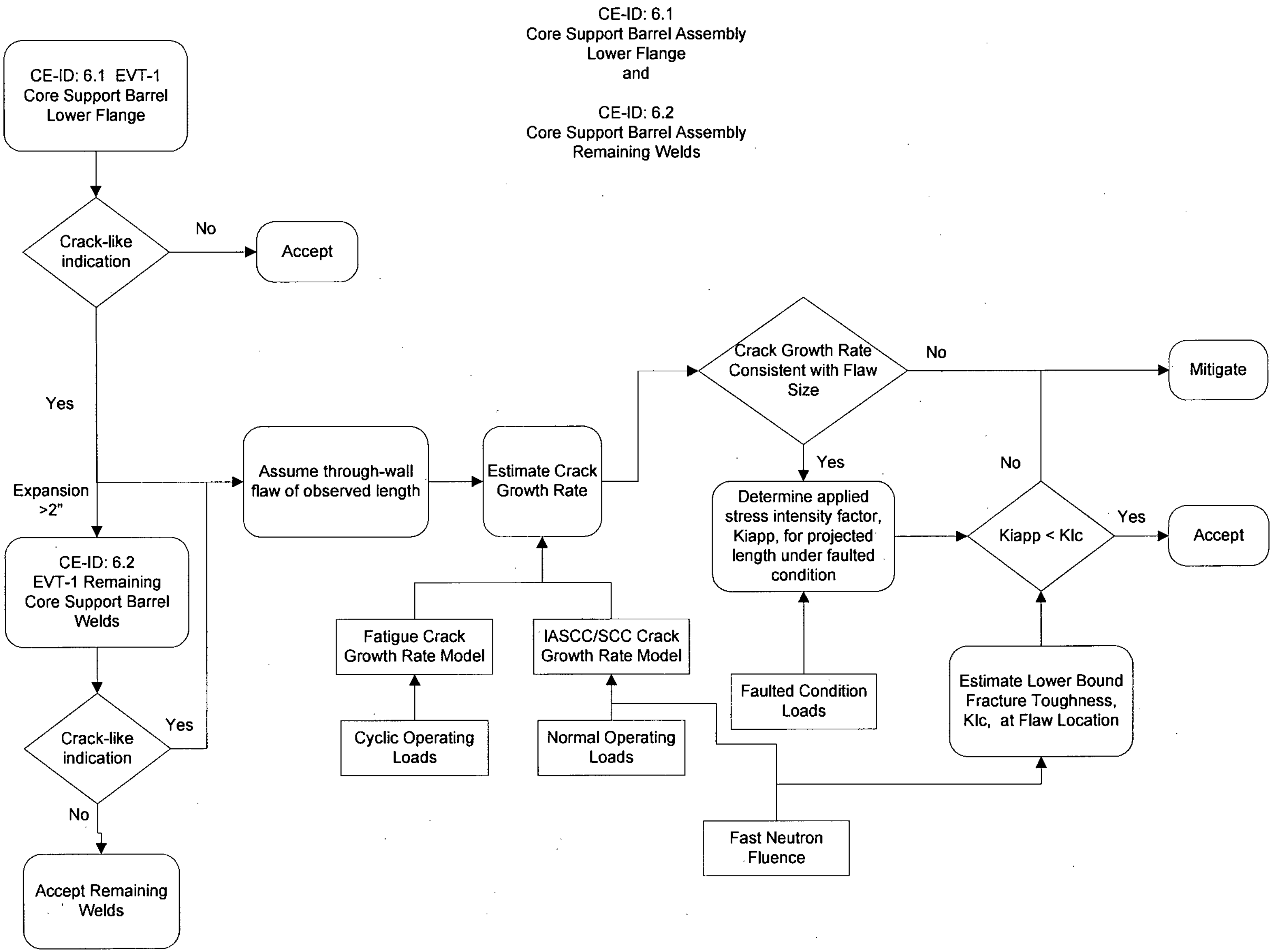


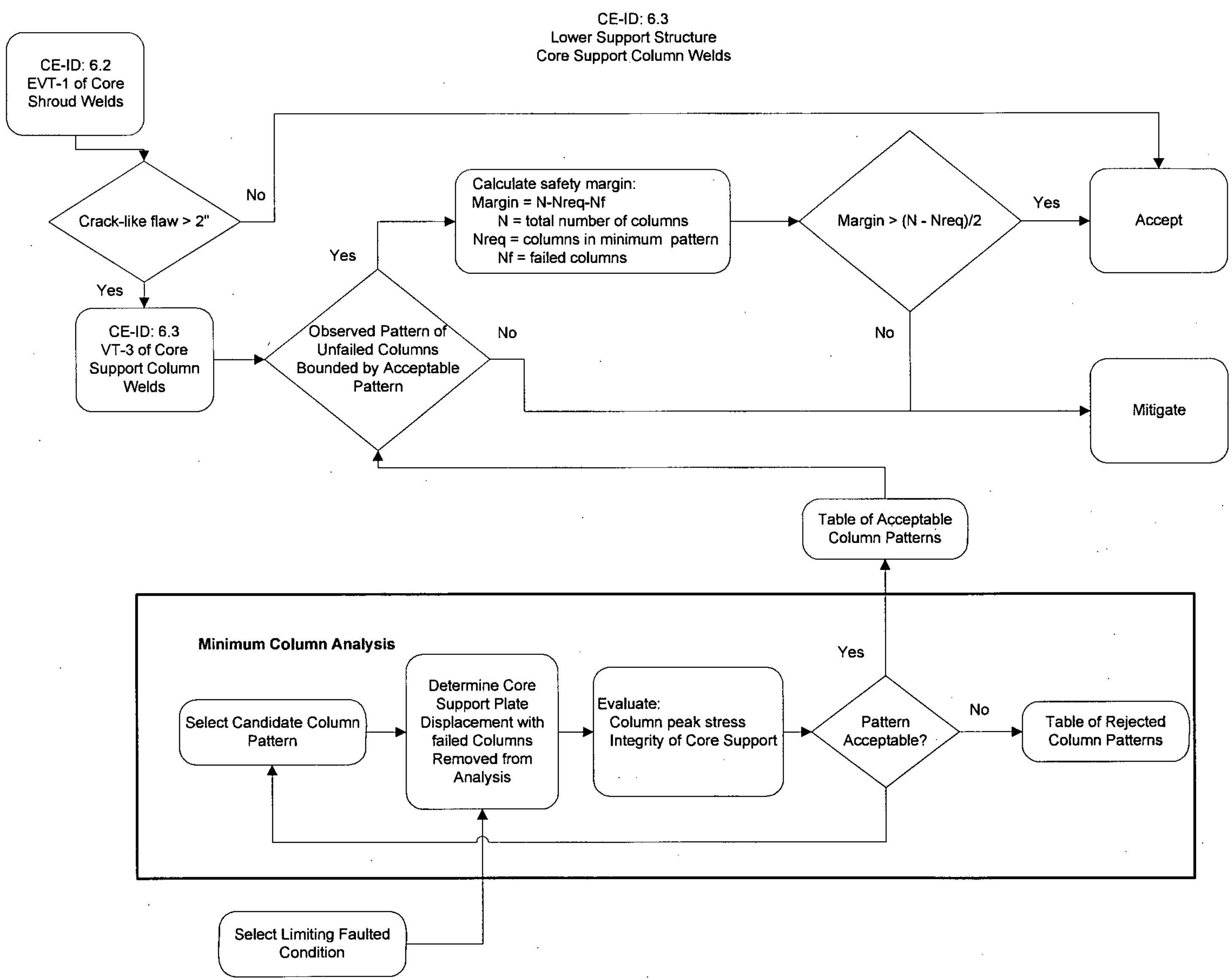
CE-ID: 6

Core Support Barrel Assembly – Upper Flange Weld

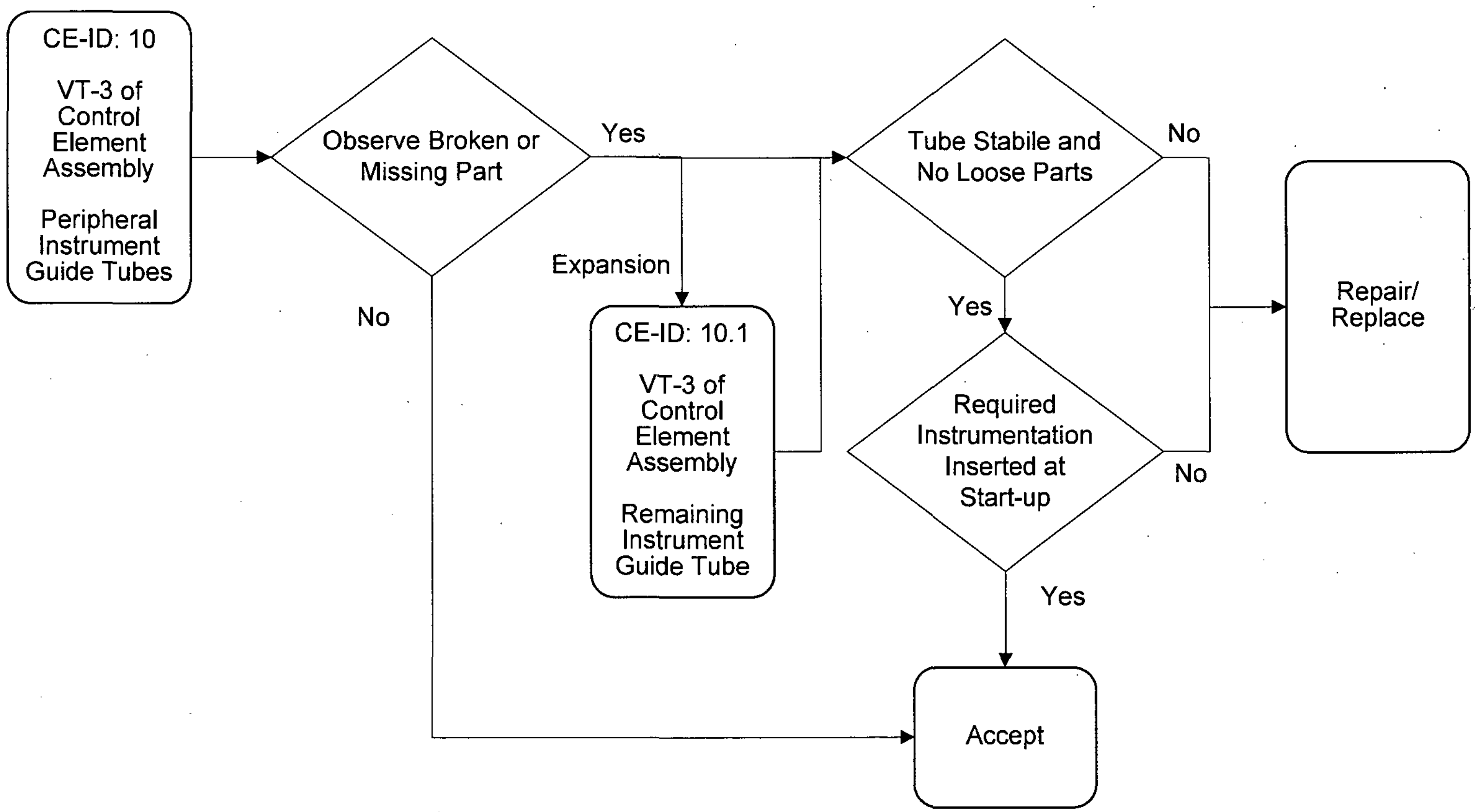
Option 2: ID Inspection




CE-ID: 6.1
Core Support Barrel Assembly
Lower Flange
and
CE-ID: 6.2
Core Support Barrel Assembly
Remaining Welds
CE-ID: 6.1 EVT-1
Core Support Barrel
Lower Flange
Crack-like indication
No
Accept
Yes
Expansion >2"
CE-ID: 6.2
EVT-1 Remaining
Core Support Barrel
Welds
Crack-like indication
Yes
No
Accept Remaining Welds
Assume through-wall flaw of observed length
Estimate Crack Growth Rate
Fatigue Crack Growth Rate Model
IASCC/SCC Crack Growth Rate Model
Cyclic Operating Loads
Normal Operating Loads
Crack Growth Rate Consistent with Flaw Size
No
Mitigate
Yes
Determine applied stress intensity factor, Kiapp, for projected length under faulted condition
No
Kiapp < KIc
Yes
Accept
Faulted Condition Loads
Estimate Lower Bound Fracture Toughness, KIc, at Flaw Location
Fast Neutron Fluence


CE-ID: 6.3
Lower Support Structure
Core Support Column Welds
CE-ID: 6.2
EVT-1 of Core
Shroud Welds
Crack-like flaw > 2"
No
Yes
CE-ID: 6.3
VT-3 of Core
Support Column
Welds
Observed Pattern of
Unfailed Columns
Bounded by Acceptable
Pattern
Yes
No
Calculate safety margin:
Margin = N-Nreq-Nf
N = total number of columns
Nreq = columns in minimum pattern
Nf = failed columns
Margin > (N - Nreq)/2
Yes
No
Accept
Mitigate
Table of Acceptable
Column Patterns
Minimum Column Analysis
Select Candidate Column
Pattern
Determine Core
Support Plate
Displacement with
failed Columns
Removed from
Analysis
Evaluate:
Column peak stress
Integrity of Core Support
Pattern
Acceptable?
Yes
No
Table of Rejected
Column Patterns
Select Limiting Faulted
Condition

CE-ID:10 &10.1
Control Element Assembly
Instrument Guide Tubes



APPENDIX E
ACCEPTANCE CRITERIA METHODOLOGY AND DATA REQUIREMENTS FOR WESTINGHOUSE COMPONENTS INCLUDED IN MRP-227

Westinghouse Primary and Expansion Components

W-ID: 1 Control Rod Guide Tube Assembly – Guide Pates (Cards)
W-ID: 2 Control Rod Guide Tube Assembly – Lower Flange Welds

- W-ID: 2.1 Lower Support Assembly – Lower Support Column Bodies (Cast)
- W-ID: 2.2 Bottom-mounted Instrumentation (BMI) System – BMI Column Bodies

W-ID: 3 Core Barrel Assembly – Upper Core Barrel Flange Weld

- W-ID: 3.1 Core Barrel Assembly – Core Barrel Flange, Core Barrel Outlet Nozzles, Lower Core Barrel Flange Weld

- W-ID: 3.2 Lower Support Assembly – Lower Support Columns (non cast)

W-ID: 4 Baffle-former Assembly – Baffle-edge Bolts
W-ID: 5 Baffle-former Assembly – Baffle-Former Bolts

- W-ID: 5.1 Core Barrel Assembly – Barrel-Former Bolts
- W-ID: 5.2 Lower Support Assembly – Lower Support Column Bolts

W-ID: 6 Baffle-Former Assembly – Assembly
W-ID: 7 Alignment and Interfacing Components – Internal Hold-down Spring
W-ID: 8 Thermal Sleeve Assembly – Thermal Shield Flexures

W-ID: 1 **Control Rod Guide Tube Assembly**

Guide Plates (Cards)

Category:	Primary Applicability: All plants
Degradation Effect:	Loss of Material (Wear)
Expansion Link:	None.
Function:	The control rod guide tube assembly provides alignment and insertion path for the control rods through the upper internals. Guide cards provide alignment and insertion path for control rod assemblies and support the control rods when withdrawn.

Inspection

Method:	Visual (VT-3) examination no later than 2 refueling outages from the beginning of the license renewal period, and no earlier than two refueling outages prior to the start of the license renewal period. Subsequent examinations are required on a 10-year interval.
Coverage:	20% examination of the number of CRGT assemblies, with all guide cards within each selected CRGT assembly examined. See MRP-227 Figure 4-20
Observable Effect:	Observation of wear requires internal visual inspections of guide tube assemblies. Westinghouse has established procedure for quantifying wear based on calibrated visual exams for the PWROG. The Westinghouse procedures meet and exceed the VT-3 requirements. The VT-3 inspections should be able to identify ligaments on inner guidance holes.

Failure

Failure Mechanism:	The guidance holes in the guide cards are distorted by wear (loss of material). Largest amounts of wear typically observed in lowest guide card levels.
Failure Effect:	Guidance hole wear can cause lack of alignment. Lack of alignment may cause a degradation of control rod drop times. In the worst case scenario, rod may jam and prevent insertion.
Failure Criteria:	Leading indicator of failure is considered to be observation of sharp tip at inner guide card slots. • Wear such that rod may escape is currently considered as failure of guide card. • Failure requires control rod to wear through ligament in guide card.

Methodology

Goal:	Rod must be restrained to guidance hole in card.
Data Requirements:	Guide card wear model Material properties Current geometry Wear trend Vertical and horizontal location Maintenance practices

W-ID: 1 **Control Rod Guide Tube Assembly**

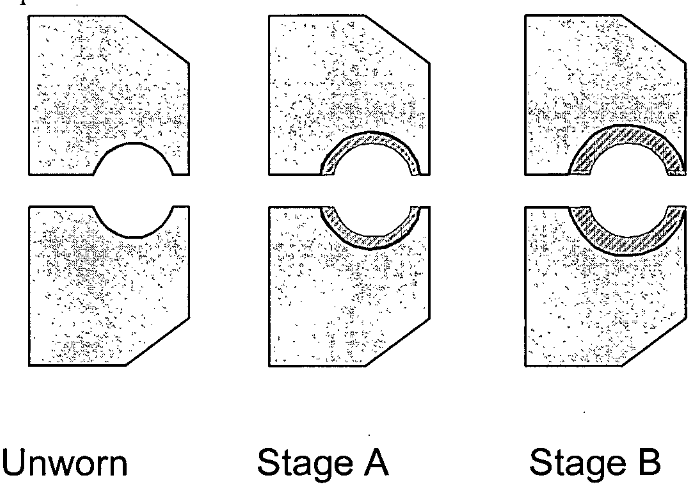
Guide Plates (Cards)

Control Rod Insertion Data (Historical)

Analysis: Two stages of wear:

A. Wear through full ligament. Can observe enlargement of guide card hole, but wear does not extend to inside surface of guide card.

B. Wear area intersects inner surface of guide card, but wear slot still too narrow to allow escape of control rod.



During Stage A, one should be able to observe a flat, unworn surface on slots in inner hole. Upon transition from Stage A to Stage B, there is no observable slot. Sharp point observed where wear area intersects inner surface of guide card.

Bounding calculation for wear life.

1. Must assume "typical" wear patterns as previously observed in PWROG program.
2. Calculate wear volume (Va) at transition from Stage A to Stage B.
3. Calculate wear volume (Vb) at point where width of wear area at guide tube inner surface is equivalent to control rod diameter.
4. Calculate fraction wear f = Va/Vb.
5. Calculate remaining wear life T = (1/f - 1)Tcur (Tcur = current operating time)

Inspection interval must be less than remaining wear life.

Acceptance Criteria: Require control rod to be captured in guide card hole.

The unworn section of guide card slot must be observable at all inner guide tube holes at each guide card level.

Demonstrate wear remains in Stage A (see analysis).

Approach: Generic work ongoing under PWROG program

Validate and/or modify linear volumetric wear rate model

Potential extension

Alternative justification that allows wear through ligament in one or more cards

W-ID: 2 **Control Rod Guide Tube Assembly**

Lower Flange Welds

Category: Primary Applicability: All plants

Degradation Effect: Cracking (SCC, fatigue)

Expansion Link: Bottom-mounted instrumentation (BMI) column bodies, Lower support column bodies (cast)

Function: The control rod guide tube assembly provides alignment and insertion path for control rods through upper internals. The lower flange welds retain the structural alignment of the component. Guide tubes must maintain rod stability in normal and LOCA transients.

Inspection

Method: Enhanced visual (EVT-1) examination to determine the presence of crack-like surface flaws in flange welds no later than 2 refueling outages from the beginning of the license renewal period and subsequent examination on a 10-year interval.

Coverage: 100% of outer (accessible) CRGT lower flange weld surfaces and adjacent base metal.

See MRP-227 Figure 4-21.

Observable Effect: Any individual weld with observed crack must be assumed to have failed.

The vertical beam portion appears to be out of position.

Failure

Failure Mechanism: Flow in the upper head applies bending moment to control rod guide tube assembly. Maximum bending stresses tend to occur near top of continuous guidance section. Stresses may lead to formation of SCC or fatigue cracks. Weld cracking may lead to loss of stiffness in guide tube assembly and loss of support capability.

Failure Effect: Loss of structural stability. Excessive deflection could impede control assembly insertion.

Failure Criteria: Design limits on the CRGT assembly are generally expressed as a maximum allowable load, which is determined based on the assembly compliance. This analysis implies a maximum allowable deflection. Interference between the guide cards and the guide tubes occur when the deflection exceeds this limit.

Methodology

Goal: Stiffness of assembly with failed welds must be sufficient to maintain allowable deflections when LOCA and SSE loads are applied. Allowable load on control rod guide tube assembly is defined by empirical testing.

Data Requirements: Loads

Finite element model of lower CRGT assembly to evaluate weld failures calibrated to benchmark data

Analysis:

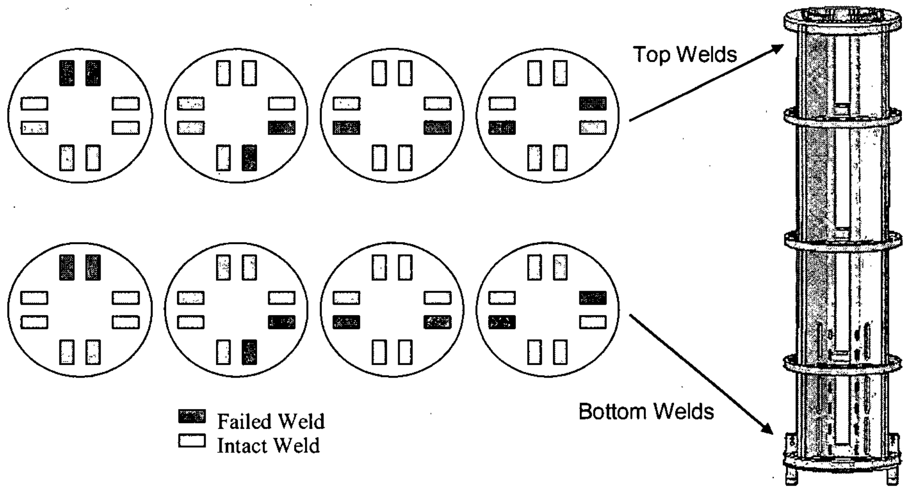
1. Determine design basis assumptions for CRGT assembly (maximum allowable load, assembly compliance).
2. Lower section must be modeled in detail, upper sections may be treated as large beams.
3. Calibrate FEA model and boundary constraints against design basis assumptions.

W-ID: 2 **Control Rod Guide Tube Assembly**

Lower Flange Welds

4. Remove test pattern of welds.
5. Run FEA.
6. If deflection is greater than limit in Step 1, pattern is not acceptable.
7. Iterate Steps 4–6 to create library of acceptable and unacceptable patterns.
8. Match patterns to field observations assuming that any weld with flaw has failed.
9. Should be able to observe sufficient number of welds to demonstrate that assembly is acceptable.

Acceptance Criteria: This acceptance criteria is based on a minimum number of welds that must continue to function (without cracking) to allow scramming of the control rods in the event of combined LOCA and SSE.



Approach: Plant-specific analysis due to large variety of sizes and designs. There may be some potential for smaller plant groupings.

W-ID: 2.1	**Lower Support Assembly**		
	Lower Support Column Bodies (Cast)		
Category:	Expansion	Applicability:	All plants
Degradation Effect:	Cracking (IASCC) including the detection of fractured support columns		
Expansion Link:	Control rod guide tube (CRGT) lower flanges		
Function:	The lower support columns provide the structural link between the lower core plate and the lower support structure. The supports are required to keep the lower core plate from deforming during operation.		

Inspection

Method: Visual (EVT-1) examination

Coverage: 100% of accessible support columns

See MRP-227 Figure 4-34.

Observable Effect: Fracture

Potential for core tilt

Control rod insertion problems

Failure

Failure Mechanism: The upper sections of the core supports may experience neutron fluences above the threshold for IASCC. The cast components are considered separately because there is a concern that they may be more sensitive to irradiation. Although stresses in columns are primarily compressive, bending stresses or the design of the attachment may produce localized regions of tensile stress.

Failure Effect: Displacement of lower core plate

Failure Criteria: Must maintain sufficient number of intact support columns to assure dimensional stability of lower core plate.

Methodology

Goal: Establish minimum acceptable pattern of core support columns. Evaluation of cast components should consider potential effect of thermal embrittlement in addition to irradiation embrittlement.

Data Requirements:

- Loads on lower core plate
- Constitutive model for stainless steel properties as a function of irradiation and thermal aging
- Displacement tolerances on lower core plate

Analysis:

1. Establish minimum functional requirements and number of core support columns to maintain structure and functional stability.
   A. During normal operation system of support columns should resist core plate deformation due to mechanical or thermal loading. Core plate requirements for "flatness" and fuel assembly alignment.
   B. During limiting accident transient system must maintain structural integrity.

W-ID: 2.1 **Lower Support Assembly**

Lower Support Column Bodies (Cast)

2. Support column analysis assumptions.
   A. Assume any column with crack in main body to have failed.
   B. Assume any column with a crack in attachment device or bolt to result in failure of the attachment.
3. Structural model of lower support structure.
   - FEA model of lower support structure that includes support columns and lower core plate. Model should be capable of removing individual column or breaking attachment to lower core plate. Would require multiple iterations to establish "minimum acceptable patterns" of core support columns and support column bolts.
4. Structural model must be run for functional requirements A and B.
5. Determine margin for additional failures.
   A. Assume number of failures in next 10 years is equal to number observed to date.

   N = # of support columns

   Nf = # of observed flawed columns

   Nreq = # of columns in relevant minimum pattern

   Margin = N-Nreq

Acceptance Criteria: Require that no more of 1/2 of columns in margin are failed:

Nf < (N-Nreq)/2

Approach: Generic program to share first-of-a-kind effort. (See W-ID: 2-1)

- Pilot analysis of lower support structure to identify critical issues.

Expect final acceptance based on plant-specific analysis.

W-ID: 2.2 **Bottom-mounted Instrumentation System**

Bottom-mounted Instrumentation (BMI) Column Bodies

Category: Expansion Applicability: All plants

Degradation Effect: Cracking (fatigue) including the detection of completely fractured column bodies

Expansion Link: Control rod guide tube (CRGT) lower flanges

Function:

- The BMI columns define the path for flux thimbles to be inserted into the fuel assemblies.
- Flux thimbles are normally withdrawn prior to refueling and re-inserted at end of refueling.
- The plant must maintain a required number of functioning flux thimbles for core mapping.

Inspection

Method: Visual (VT-3) examination of BMI column bodies as indicated by difficulty of insertion/withdrawal of flux thimbles. Flux thimble insertion/withdrawal to be monitored at each inspection interval.

Coverage: 100% of BMI column bodies for which difficulty is detected during flux thimble insertion/withdrawal.

See MRP-227 Figure 4-35.

Observable Effect:

- Fracture should be readily visible
- Large loose parts
- Skewed flow
- Weakened support

Failure

Failure Mechanism: The BMI columns may be subject to fatigue due to either thermal fatigue or flow induced vibrations.

Failure Effect: Inability to insert flux thimbles. This effect would be noted during refueling outage. Consequences of failure during ensuing operating period are believed to be minimal.

Failure Criteria:

- The plant must maintain a required number of functioning flux thimbles for core mapping.
- Any BMI column with an observable crack will be assumed to have failed.
- The primary pressure boundary must be intact.

Methodology

Goal: Configuration of unfailed BMI columns should be sufficient to allow required flux mapping. (Installation of WINCISE™ may obviate the need for the entire BMI system.)

Data Requirements: Criteria should be part of plant technical specifications.

Analysis: Evaluate stability of failed BMI Column. Any section that could potentially detach and become a loose part or otherwise interfere with plant operation should be removed or stabilized.

W-ID: 2.2 **Bottom-mounted Instrumentation System**

Bottom-mounted Instrumentation (BMI) Column Bodies

Acceptance Criteria: Plant must have minimum number of unfailed BMI assemblies to allow flux-mapping at startup.

Approach: Pass/Fail inspection with established minimum number of instrumentation tubes. Based directly on plant specifications.

W-ID: 3	**Core Barrel Assembly**	
	Upper Core Barrel Flange Weld	
Category:	Primary	Applicability: All plants
Degradation Effect:	Cracking (SCC)	
Expansion Link:	Remaining core barrel welds (core barrel flange, core barrel outlet nozzles, lower core barrel flange weld), lower support column bodies (non cast)	
Function:	Primary core support structure.	

Inspection

Method: Periodic enhanced visual (EVT-1) examination, no later than 2 refueling outages from the beginning of the license renewal period and subsequent examination on a 10-year interval.

Coverage: 100% of one side of the accessible surfaces of the selected weld and adjacent base metal.

See MRP-227 Figure 4-22.

Observable Effect: Stress corrosion crack along seam weld.

Failure

Failure Mechanism: SCC

Failure Effect: Potential loss of core support.

Failure Criteria: Actively growing through-wall flaws require mitigation. Require demonstration that flaw growth is arrested or limited to surface.

An existing through-wall flaw may be acceptable if condition and shape indicate that it is a non-growing fabrication flaw.

Methodology

Goal: Due to the high fracture toughness of unirradiated stainless steel, the core barrel is a highly flaw tolerant structure and flaw sizes are expected to be very large. However, the core barrel is a critical support structure. Flaw growth in this component is outside the range of normal expectations. Therefore, it has been assumed that the presence of any actively growing through-wall crack would require repair or other mitigation. The goal of the calculation is to demonstrate the crack is stable or not likely to grow through wall.

Data Requirements:

1. Operating loads
2. K solutions for range of expected crack shapes (lengths and depths)
3. SCC crack growth rate curves
4. Fatigue crack growth rate curve (as backup)

Analysis:

Option 1. Observation on OD of core barrel

Step 1. Determine stress distribution through core barrel thickness for normal operating conditions (expect peak stress at vessel OD).

Step 2. Obtain stress intensity factor solution for part-through-wall crack as function of surface length (L) and depth (a).

Step 3. Short cracks will be constrained by the stress distribution in the

W-ID: 3 **Core Barrel Assembly**

Upper Core Barrel Flange Weld

barrel wall. Define the maximum constrained crack length as Lc.

Step 4. OD crack observation is acceptable if L < Lc.

Step 5 If L > Lc, then must perform UT to determine crack depth (a).

Step 6. Crack is acceptable if K corresponding to a and Lc is less than 20 ksi-in^1/2.

Step 7. All remaining cracks require specific flaw-tolerance analysis.

Option 2. Observation of flaw on ID of core support barrel

Step 1. If flaw on ID is smaller than the length (Lc) defined in Option 1, visually examine the OD surface corresponding to the ID flaw to determine if it is OD-initiated. Crack is acceptable if not through-wall.

Step 2. For a through-wall flaw, apply the OD flaw acceptance criteria from Option 1.

Step 3. All remaining cracks require a geometry-specific flaw-tolerance analysis.

Option 3. Observation of crack on ID of core support barrel

Step 1. If flaw on ID is smaller than the length (Lc) defined in Option 1, perform UT exam to determine if the crack is through-wall. Crack is acceptable if not through-wall.

Step 2. For a through-wall flaw, apply the OD flaw acceptance criteria from Option 1.

Step 3. All remaining cracks require a geometry-specific flaw-tolerance analysis.

Acceptance Criteria: Demonstrate that crack is not actively growing or limited to surface as indicted by analysis.

Approach: Plant-specific analysis.

- Ginna provides pilot plant experience for the creation of generic acceptance criteria.

May be able to group plants by design.

W-ID: 3.1 Core Barrel Assembly

Core Barrel Flange, Core Barrel Outlet Nozzles, Lower Core Barrel Flange Weld

Category:	Expansion Applicability: All plants
Degradation Effect:	Cracking (SCC, fatigue)
Expansion Link:	Upper core barrel flange weld
Function:	Primary core support structure

Inspection

Method:	Enhanced visual (EVT-1) examination, with initial examination and re-examination frequency dependent on the examination results for upper core barrel flange
Coverage:	100% of one side of the accessible surfaces of the selected weld and adjacent base metal See MRP-227 Figure 4-22.
Observable Effect:	Cracking along line of weld.

Failure

Failure Mechanism:	SCC, fatigue
Failure Effect:	Potential loss of core support
Failure Criteria:	Actively growing through-wall flaws require mitigation. Require determination of crack growth mechanism. An existing through-wall flaw may be acceptable if condition and shape indicate that it is a non-growing fabrication flaw.

Methodology

Goal: Demonstrate that cracking mechanism is understood and projected crack growth is limited.

Data Requirements:

1. Operating loads
2. K solutions for range of expected crack shapes (lengths and depths)
3. SCC crack growth rate curves
4. Fatigue crack growth rate curve (as backup)

Analysis:

1. Flaws in core barrel above the baffle section will be evaluated assuming active crack growth mechanisms are SCC and fatigue.
2. Flaws in the beltline region of the core barrel (care baffle section) will be evaluated assuming active growth mechanisms are IASCC and fatigue.
3. A fluence estimate at the flaw location is required for all flaws in the beltline region.
4. Normal operating and fatigue loads will be established for core barrel at this location.
5. Determine stress intensity factors for a through-wall crack.
6. Use appropriate crack growth rate models (SCC or IASCC and fatigue) to estimate crack growth rate.

W-ID: 3.1

Core Barrel Assembly

Core Barrel Flange, Core Barrel Outlet Nozzles, Lower Core Barrel Flange Weld

7. If crack growth rate is consistent with observed flaw size:
    - Project flaw size through inspection interval using crack growth rate estimate.
    - Determine loads during limiting transient.
    - Determine stress intensity factor for through-wall crack of projected length
    - For low fluence region assume K_{Ic} = 150 ksi-in^1/2
    - For beltline region determine lower bound toughness based on fluence estimate.
    - If stress intensity factor during transient is less than fracture toughness, flaw is acceptable.
    - If stress intensity factor during transient is greater than fracture toughness, proceed to Step 8
8. If crack growth rate is too low to explain existence of observed crack or flaw not acceptable by Step 7:

Determine crack depth

- If crack depth small compared to barrel thickness (< xx inches) then crack is acceptable.
- If crack depth large compared to barrel thickness, the crack is rapidly growing and a detailed analysis is required.

Acceptance Criteria: Current crack size is explainable by known crack growth rate laws and limited crack growth is projected.

Approach: Plant-specific analysis.

- Require flaw tolerance handbook/methodology based on flaw location and direction.

MRP-210 may have limited relevance.

W-ID: 3.2 **Lower Support Assembly**

Lower Support Column Bodies (Non Cast)

Category: Expansion Applicability: All plants

Degradation Effect: Cracking (IASCC)

Expansion Link: Upper core barrel flange weld

Function: The lower support columns provide the structural link between the lower core plate that supports the fuel assemblies and the relatively thick lower support forging (or in a limited number of cases casting.) The supports are required to keep the lower core plate from deforming during operation.

Inspection

Method: Enhanced visual (EVT-1) examination, with initial examination and re-examination frequency dependent on the examination results for upper core barrel flange weld.

Coverage: 100% of accessible surfaces

See MRP-227 Figure 4-34.

Observable Effect:

- Fracture
- Potential for core tilt
- Control rod insertion problems

Failure

Failure Mechanism: The upper sections of the core supports may experience neutron fluences above the threshold for IASCC. Although the main stresses in the support is expected to be compressive, bending stresses or the design of the attachment may produce localized regions of tensile stress.

Failure Effect: Displacement of lower core plate

Failure Criteria: Must maintain sufficient number of intact support columns to assure dimensional stability of lower core plate.

Methodology

Goal: Establish minimum acceptable pattern of core support columns.

Data Requirements: Loads on lower core plate

Constitutive model for stainless steel properties as a function of irradiation and thermal aging.

Displacement tolerances on lower core plate

Analysis:

1. Establish minimum functional requirements and number of core support columns to maintain structure and functional stability.
   A. During normal operation system of support columns should resist core plate deformation due to mechanical or thermal loading. Core plate requirements for "flatness" and fuel assembly alignment.
   B. During limiting accident transient system must maintain structural integrity.

W-ID: 3.2 **Lower Support Assembly**

Lower Support Column Bodies (Non Cast)

2. Support column analysis assumptions.
   A. Assume any column with crack in main body to have failed.
   B. Assume any column with a crack in attachment device or bolt to result in failure of the attachment.
3. Structural model of lower support structure.
   - FEA model of lower support structure that includes support columns and lower core plate. Model should be capable of removing individual column or breaking attachment to lower core plate. Would require multiple iterations to establish "minimum acceptable patterns" of core support columns and support column bolts.
4. Structural model must be run for functional requirements A and B.
5. Determine margin for additional failures.

A. Assume number of failures in next 10 years is equal to number observed to date.

N = # of Support Columns

Nf = # of Observed Flawed Columns

Nreq = # of columns in relevant minimum pattern

$$\text{Margin} = N - N_{req}$$

Acceptance Criteria: Require that no more of 1/2 of columns in margin are failed:

$$N_f < (N - N_{req})/2$$

Approach: Generic program to share first-of-a-kind effort. (See W-ID: 2-1)

- Pilot analysis of lower support structure to identify critical issues.
- Expect final acceptance based on plant-specific analysis.

W-ID: 4	**Baffle-former Assembly**		
	Baffle-edge Bolts		
Category:	Primary	Applicability:	All plants with baffle-edge bolts
Degradation Effect:	Cracking (IASCC, fatigue) that results in		
Expansion Link:	None		
Function:	The baffle-edge bolts provide the baffle-plate to baffle plate attachment along the seam between plates. The edge bolts prevent gaps between plants that can result in baffle-jetting damage to peripheral fuel assemblies. Studies have demonstrated that baffle edge bolts are not required to maintain the structural integrity of the baffle.		

Inspection

Method:	Visual (VT-3) examination, with baseline examination between 20 and 40 EFPY and subsequent examinations on a 10-year interval.
Coverage:	Bolts and locking devices on high fluence seams. 100% of components accessible from core side. See MRP-227 Figure 4-23.
Observable Effect:	Failure of bolt or locking device as listed under inspection.

Failure

Failure Mechanism:	Analysis has shown that differential thermal expansion and swelling can cause plastic deformation of edge bolts. These bolts are in high radiation locations and there is a significant potential failure due to IASCC. Failure modes considered should include: • Broken or missing locking devices • Protruding bolt heads • Missing bolts or bolt heads
Failure Effect:	In plants with downward coolant flow in the region between the baffle and the former, failure may contribute to baffle jetting. Primary concerns are loose parts generation and interference with fuel.
Failure Criteria:	All bolts and locking devices should be in place and undamaged. FMEA should be completed prior to analysis to identify potential observations. Pre-planned responses to be implemented.

Methodology

Goal:	A plant-specific plan should be developed for evaluating and mitigating the potential relevant conditions. The evaluation should consider any previously reported observations.
Data Requirements:	FMEA results
Analysis:	Prepare for examination by conducting a failure modes and effects analysis (FMEA) to identify full range of potential relevant observations prior to inspection.

W-ID: 4 **Baffle-former Assembly**

Baffle-edge Bolts

Acceptance Criteria: Determined by FMEA

Approach: FMEA should address plant-specific practices and priorities. Some generic work possible to outline issues and options to be addressed in FMEA.

W-ID: 5 **Baffle-former Assembly**

Baffle-former Bolts

Category:	Primary	Applicability:	All plants
Degradation Effect:	Cracking (IASCC, fatigue)		
Expansion Link:	Lower support column bolts, barrel-former bolts		
Function:	The baffle-former bolts attach the baffle plates to the formers.		

Inspection

Method: Baseline volumetric (UT) examination between 25 and 35 EFPY, with subsequent examination after 10 to 15 additional EFPY to confirm stability of bolting pattern. Re-examination for high-leakage core designs requires continuing examinations on a 10-year interval.

Coverage: 100% of accessible bolts or as supported by plant-specific justification. Heads accessible from the core side. UTaccessibility may be affected by complexity of head and locking device designs.

See MRP-227 Figures 4-23 and 4-24.

Observable Effect: UT will detect bolts with large cracks (approx. 30%) through of cross-sectional area. Fractured bolts should be captured by locking devices – no visible indication.

Failure

Failure Mechanism: Known IASCC cracking of similar highly irradiated bolts has been reported.

Failure Effect: Loss of structural stability

Failure Criteria: Require a minimum bolting pattern

Methodology

Goal: Must demonstrate that projected number of additional bolt failures will not threaten minimum pattern prior to next scheduled inspection.

Data Requirements:

- Loads
- Bolting patterns
- Baffle design
- Fast neutron (dpa) distribution in core shroud
- Projected bolt failure rate
- Minimum bolting pattern analysis

Analysis: The observed pattern of failed bolts must meet the pre-defined acceptable bolt pattern and have a reasonable margin to protect against additional failures during the inspection interval. The margin is defined in terms of the number of intact bolts beyond the number required for the minimum bolting pattern. The margin (M) at any time is simply:

$$M = N - Nreq - Nf$$

where

N = total number of baffle-former bolts

Nreq = number of baffle-former bolts in minimum acceptable pattern

Nf = number of failed bolts

W-ID: 5 **Baffle-former Assembly**

Baffle-former Bolts

Assuming that there are no failed bolts at the beginning of life, the initial margin is simply: (N - Nreq). For operation through the next 10–15 EFPY interval, require that no more than 50% of initial margin be consumed at the time of the first inspection.

Acceptance Criteria:

1. Observed pattern of unfailed bolts meets pre-defined acceptance criteria
2. Less than 50% of initial margin consumed

$$Nf < (N - Nreq)/2$$

Approach: Generic work completed in previous PWROG program

W-ID: 5.1	**Core Barrel Assembly**
	Barrel-former Bolts
Category:	Expansion Applicability: All plants
Degradation Effect:	Cracking (IASCC, fatigue)
Expansion Link:	Baffle-former bolts
Function:	Maintain structural integrity of baffle-former-barrel structure.

Inspection

Method:	Volumetric (UT) examination, with initial and subsequent examinations dependent on results of baffle-former bolt examinations.
Coverage:	100% of accessible bolts. Accessibility may be limited by presence of thermal shields or neutron pads. See MRP-227 Figure 4-23.
Observable Effect:	UT will detect bolts with large cracks (approx. 30%) through the cross sectional area

Failure

Failure Mechanism:	Cracking Loss of bolt pre-load due to irradiation induced stress relaxation may exacerbate fatigue issue in aging plants
Failure Effect:	Potential for flow induced vibration due to loss of bolting constraint. Loss of structural stability
Failure Criteria:	UT indications

Methodology

Goal:	Must demonstrate a minimum bolting pattern.
Data Requirements:	• Loads/displacements • Bolting patterns • Baffle-former = barrel design • Fast neutron (dpa) distribution in core barrel • Projected bolt failure rate • Minimum bolting pattern analysis
Analysis:	Procedures for establishing acceptable bolting patterns for the barrel-to-former bolts have been established in [13]. This methodology has been reviewed and accepted by the NRC in a Safety Evaluation issued in 1998 (TAC No. MA1152). The PWROG has developed minimum acceptable bolting patterns for all Westinghouse designed plants in the United States. In some cases, a plant-specific bolting pattern evaluation may produce a less restrictive result.

W-ID: 5.1 **Core Barrel Assembly**

Barrel-former Bolts

The observed pattern of failed bolts must meet the pre-defined acceptable bolt pattern and have a reasonable margin to protect against additional failures during the inspection interval. The margin is defined in terms of the number of intact bolts beyond the number required for the minimum bolting pattern. The margin (M) at any time is simply:

$$M = N - Nreq - Nf$$

where

N = total number of barrel-former bolts

Nreq = number of barrel-former bolts in minimum acceptable pattern

Nf = number of failed bolts

Assuming that there are no failed bolts at the beginning of life, the initial margin is simply: (N - Nreq). For operation through the next 10–15 EFPY interval, require that no more than 50% of initial margin be consumed at the time of the first inspection.

Acceptance Criteria:

1. Observed pattern of unfailed bolts meets pre-defined acceptance criteria
2. Less than 50% of initial margin consumed

$$Nf < (N - Nreq)/2$$

Approach: Generic work completed in previous PWROG program

W-ID: 5.2	**Lower Support Assembly**		
	Lower Support Column Bolts		
Category:	Expansion	Applicability:	All plants
Degradation Effect:	Cracking (IASCC, fatigue)		
Expansion Link:	Baffle-former bolts		
Function:	The lower support column bolts attach the support columns to the lower core plate. Although the bolts do not directly support the weight of the core, they help maintain the flatness and integrity of the lower support plate.		

Inspection

Method: Volumetric (UT) examination, with initial and subsequent examinations dependent on results of baffle-former bolt examinations.

Coverage: 100% of accessible bolts or as supported by plant-specific justification.

See MRP-227 Figures 4-32 and 4-33.

Observable Effect: Failed UT inspection

Failure

Failure Mechanism: Cracking

Failure Effect: Displacement of lower core plate

Failure Criteria: Assume failure of bolt results in loss of attachment between support column and lower core plate.

Methodology

Goal: Establish functional requirements for core support columns.

A. During normal operation system of support columns should resist core plate deformation due to mechanical or thermal loading. Core plate requirements for "flatness" and fuel assembly alignment.

B. During limiting accident transient system must maintain structural integrity.

Data Requirements:

- Loads on lower core plate
- Displacement tolerances on lower core plate

Analysis:

1. Establish functional requirements for core support columns.

   A. During normal operation system of support columns should resist core plate deformation due to mechanical or thermal loading. Core plate requirements for "flatness" and fuel assembly alignment.

   B. During limiting accident transient system must maintain structural integrity.

2. Structural model of lower support structure.

   FEA model of lower support structure that includes support columns and lower core plate. Model should be capable of removing individual column or breaking attachment to lower core plate. Would require multiple iterations to establish "minimum acceptable patterns" of core support columns and support column bolts.

3. Structural model must be run for functional requirements A and B.

W-ID: 5.2 **Lower Support Assembly**

Lower Support Column Bolts

4. Determine margin for additional failures.

   A. Assume number of failures in next 10 years is equal to number observed to date.

      N = # of Support Columns

      Nf = # of Observed Flawed Columns

      Nreq = # of columns in relevant minimum pattern

      Margin = N - Nreq

Acceptance Criteria: Nf < (N - Nreq)/2

Approach: Generic program to share first-of-a-kind effort. (See W-ID: 2-1)

- Pilot analysis of lower support structure to identify critical issues.

Expect final acceptance based on plant-specific analysis.

W-ID: 6	**Baffle-former Assembly**	
	Assembly	
Category:	Primary	Applicability: All plants
Degradation Effect:	Distortion (void swelling), or cracking (IASCC) that results in	
Expansion Link:	None	
Function:	• Provide support, guidance, and protection for the reactor core • Provide a passageway for the distribution of the reactor coolant flow to the reactor core • Provide gamma and neutron shielding for the reactor vessel	

Inspection

Method:	Visual (VT-3) examination to check for evidence of distortion, with baseline examination between 20 and 40 EFPY and subsequent examinations on a 10-year interval.
Coverage:	Core side surface as indicated See MRP-227 Figures 4-24, 4-25, 4-26, and 4-27.
Observable Effect:	Degradation of general condition as described above

Failure

Failure Mechanism:	Void swelling, IASCC
Failure Effect:	1. Interference with fuel assemblies 2. Obstruction of coolant flow 3. Loose parts generation 4. Distortion/misalignment of core 5. Local temperature peaks 6. Degradation of control rod insertability 7. Baffle jetting
Failure Criteria:	No relevant observations

Methodology

Goal:	A plant-specific plan should be developed for evaluating and mitigating the potential relevant conditions. The evaluation should consider any previously reported observations.
Data Requirements:	1. Baseline data on previous visual examinations of baffle-former assembly 2. Loose parts monitoring data
Analysis:	Prepare for examination by conducting a failure modes and effects analysis (FMEA) to identify full range of potential relevant observations prior to inspection. Failure mechanisms considered should include: • Broken or missing locking devices • Protruding bolt heads • Missing bolts or bolt heads

W-ID: 6 **Baffle-former Assembly**

Assembly

- Distortion or displacement of baffle plates
- Gross cracking of baffle plates
- Gaps at plate joints
- Interaction with fuel assemblies
- Historical record

Acceptance Criteria: Determined by FMEA

Approach: FMEA should address plant-specific practices and priorities. Some generic work possible to outline issues and options to be addressed in FMEA.

W-ID: 7	**Alignment and Interfacing Components**		
	Internals Hold-down Spring		
Category:	Primary	Applicability:	All plants with 304 stainless steel hold-down springs
Degradation Effect:	Stress Relaxation		
Expansion Link:	None		
Function:	Provide hold-down forces for core internals. Retain internals in proper alignment to the core.		

Inspection

Method: Direct measurement of spring height within three cycles of the beginning of the license renewal period. If the first set of measurements is not sufficient to determine life, spring height measurements must be taken during the next two outages to extrapolate the expected spring height to 60 years.

Coverage: Measurements should be taken at several points around the circumference of the spring, with a statistically adequate number of measurements at each point to minimize uncertainty.

See MRP-227 Figure 4-28.

Observable Effect: Reduced height of core hold-down spring. Repeated measurements should indicate progressive reduction in height from cycle-to-cycle. Wear surfaces may also exhibit evidence of galling.

Failure

Failure Mechanism: Stress relaxation

Failure Effect:

- Loss of hold-down forces may lead to vibration and wear in lower internals
- Long term stress relaxation

Failure Criteria: Failure to maintain hold-down force through next inspection cycle.

Methodology

Goal: Remaining spring force must meet requirements for core hold-down forces.

Data Requirements:

- Historical information on spring height (project rate of relaxation)
- Effective spring constant
- Necessary hold-down force (plant specific)
- Current spring height
- Degradation (trending)

Analysis: Need to construct creep-stress relaxation model to define bounding (high relaxation) behavior.

- Material properties (stiffness, creep)
- History (transients, creep)
- Geometry
- Force profile

W-ID: 7	**Alignment and Interfacing Components**
	Internals Hold-down Spring
Acceptance Criteria:	Relaxation of hold-down spring must be above bounding prediction.
	Projection to end of inspection interval must assure that hold-down force maintained through next inspection interval.
Approach:	Value determined by plant-specific design requirements.

W-ID: 8 **Thermal Shield Assembly**

Thermal Shield Flexures

Category: Primary Applicability: All plants with thermal shields

Degradation Effect: Cracking (fatigue)

Expansion Link: None

Function: The flexure is the lower structural support for the thermal shield. Flexures hold the thermal shield concentric to the core. The flexure design allows for differential thermal expansion between the core barrel and the thermal shield.

Inspection

Method: Visual (VT-3) no later than 2 refueling outages from the beginning of the license renewal period. Subsequent examinations on a 10-year interval.

Coverage: 100% of thermal shield flexures

See MRP-227 Figures 4-29 and 4-36.

Observable Effect: Crack, displacement, fracture, or component separation.

Failure along weld at base of flexure or failure of weld attachment to thermal shield.

Failure

Failure Mechanism: Large deflections of the flexure due to thermal cycling may lead to fatigue failures.

Failure Effect: Failure of flexures contributes to vibration of the thermal shield. Failure can also result in flow blockage, wear, and damage to specimen guides.

Failure Criteria: Number of unfailed thermal shield flexures must be sufficient to retain structural functionality of the entire thermal shield assembly.

Methodology

Goal: Determine the number and location of thermal shield flexures that must remain intact to retain structural functionality of the entire thermal shield assembly.

Data Requirements:

- Load
- Geometry
- History (transients)
- Materials

Analysis: Perform structural assessment to determine the minimum number of flexures required to retain structural integrity.

The dynamic response of the thermal shield should be established.

Assume:

1. Any thermal shield flexure with an observed flaw has failed.
2. No credit for "bumpers" and other redundant structures.

Acceptance Criteria: Failure of a thermal shield flexure is acceptable if it can be demonstrated that the dynamic response of the thermal shield is unchanged when the flexure is removed from the model.

Any observation of a failed thermal shield flexure should lead to enhanced vigilance for fatigue and vibration monitoring systems.

Approach: Plant-specific analysis.

APPENDIX F
FLOW CHARTS OF ILLUSTRATING EVALUATION METHODOLOGIES FOR WESTINGHOUSE-DESIGNED PLANTS

Westinghouse Primary and Expansion Components

W-ID: 1 Control Rod Guide Tube Assembly – Guide Pates (Cards)
W-ID: 2 Control Rod Guide Tube Assembly – Lower Flange Welds

- W-ID: 2.1 Lower Support Assembly – Lower Support Column Bodies (Cast)
- W-ID: 2.2 Bottom Mounted Instrumentation (BMI) System – BMI Column Bodies

W-ID: 3 Core Barrel Assembly – Upper Core Barrel Flange Weld

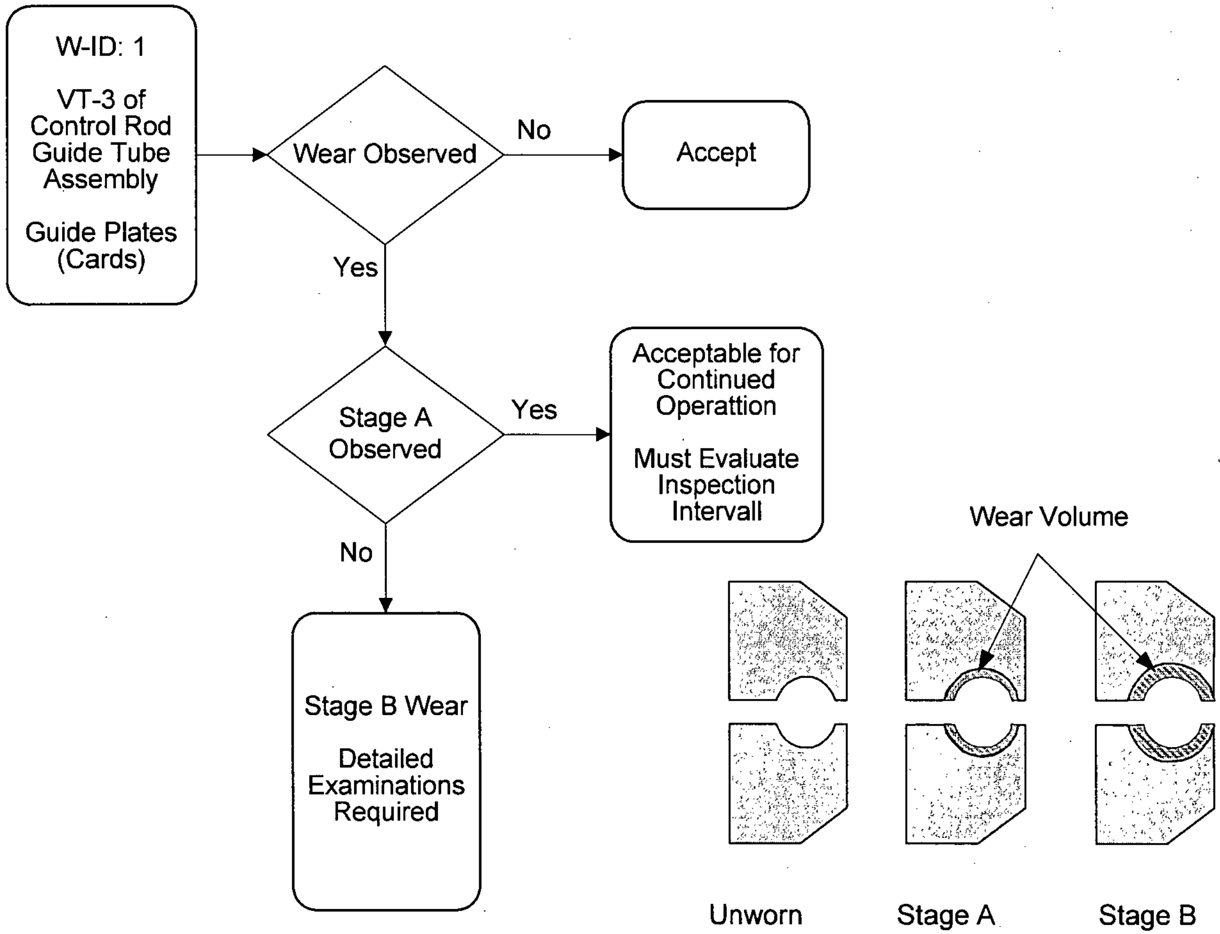
- W-ID: 3.1 Core Barrel Assembly – Core Barrel Flange, Core Barrel Outlet Nozzles, Lower Core Barrel Flange Weld
- W-ID: 3.2 Lower Support Assembly – Lower Support Columns (non cast)

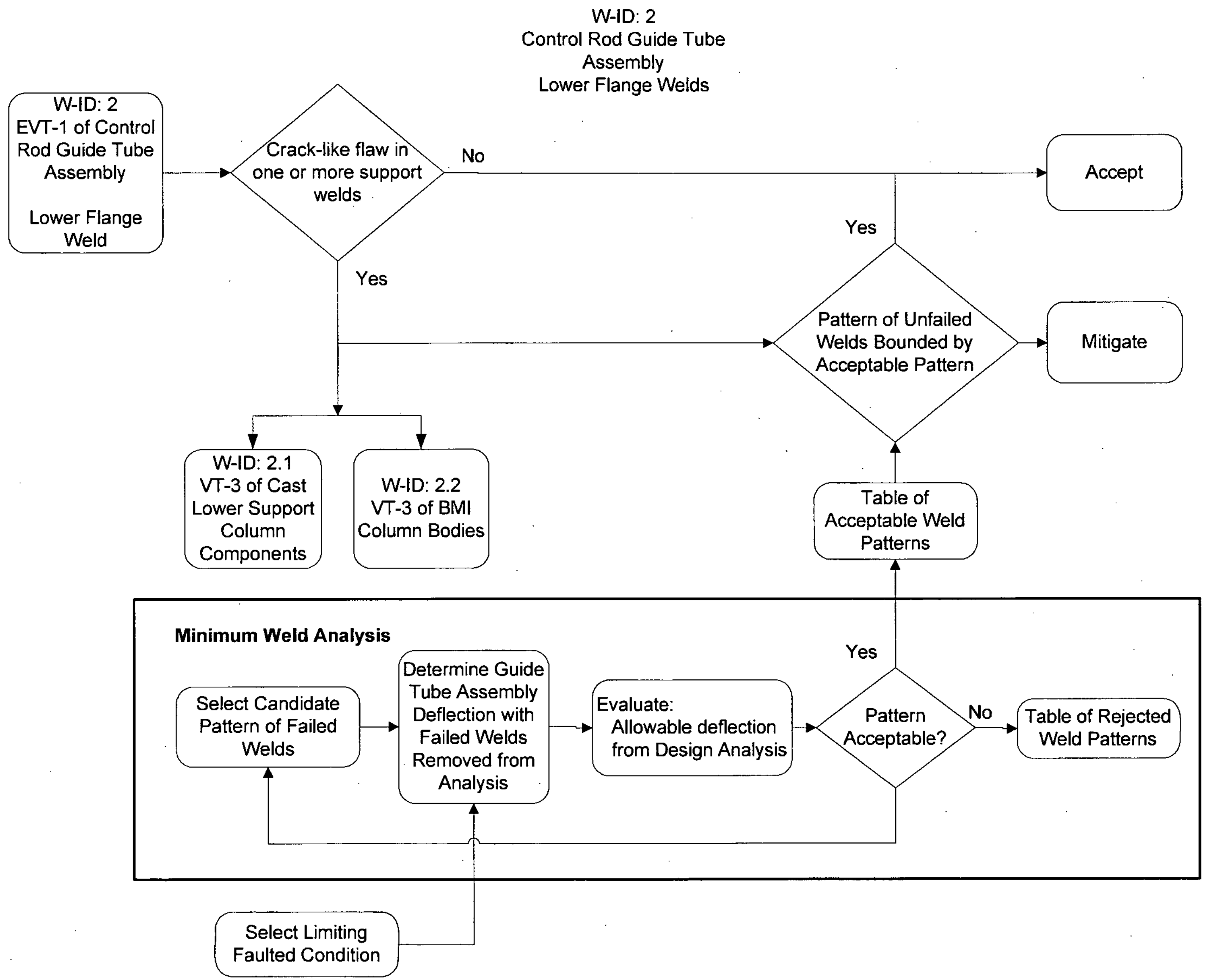
W-ID: 4 Baffle-Former Assembly – Baffle-Edge Bolts
W-ID: 5 Baffle-Former Assembly – Baffle-Former Bolts

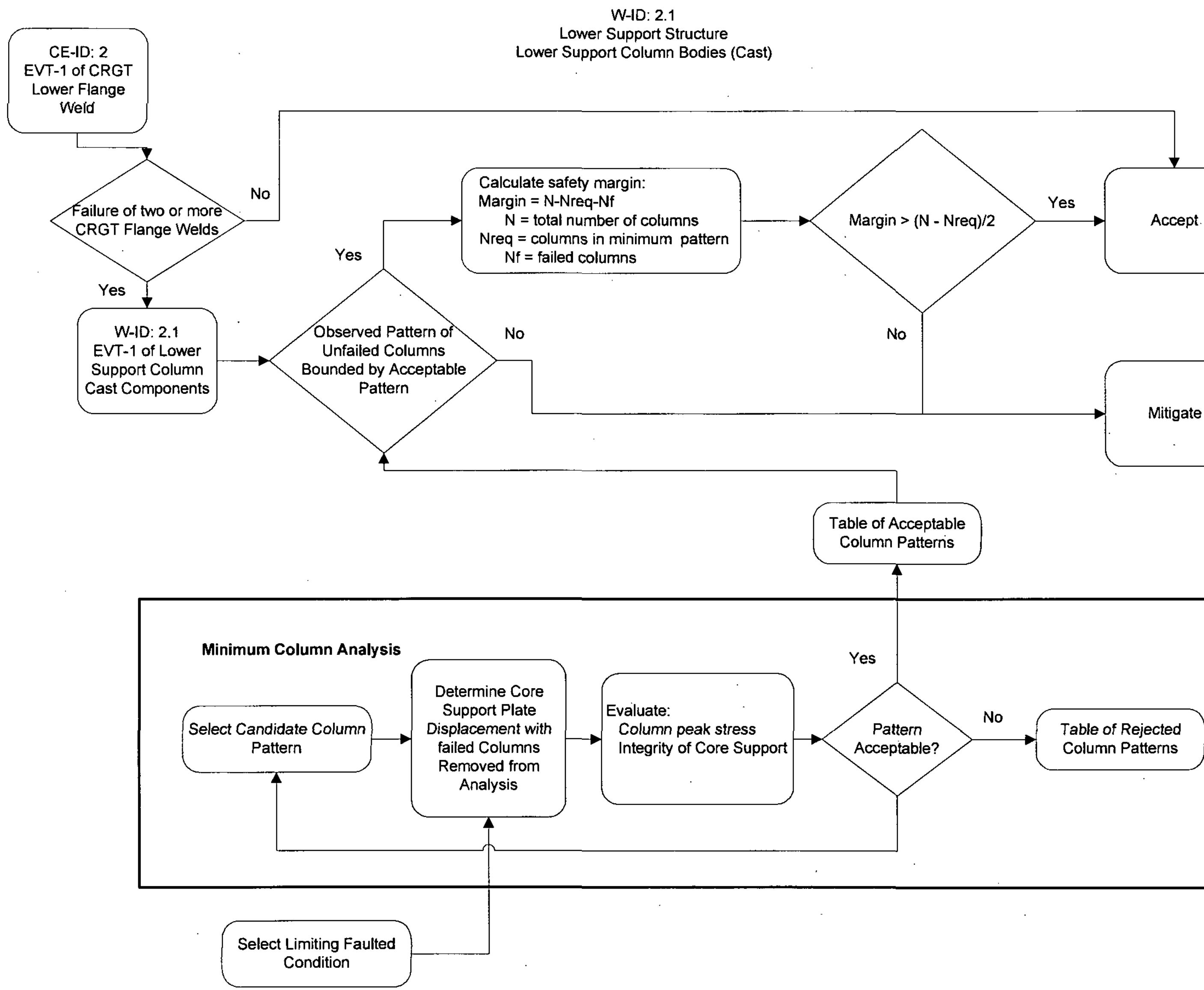
- W-ID: 5.1 Core Barrel Assembly – Barrel-Former Bolts
- W-ID: 5.2 Lower Support Assembly – Lower Support Column Bolts

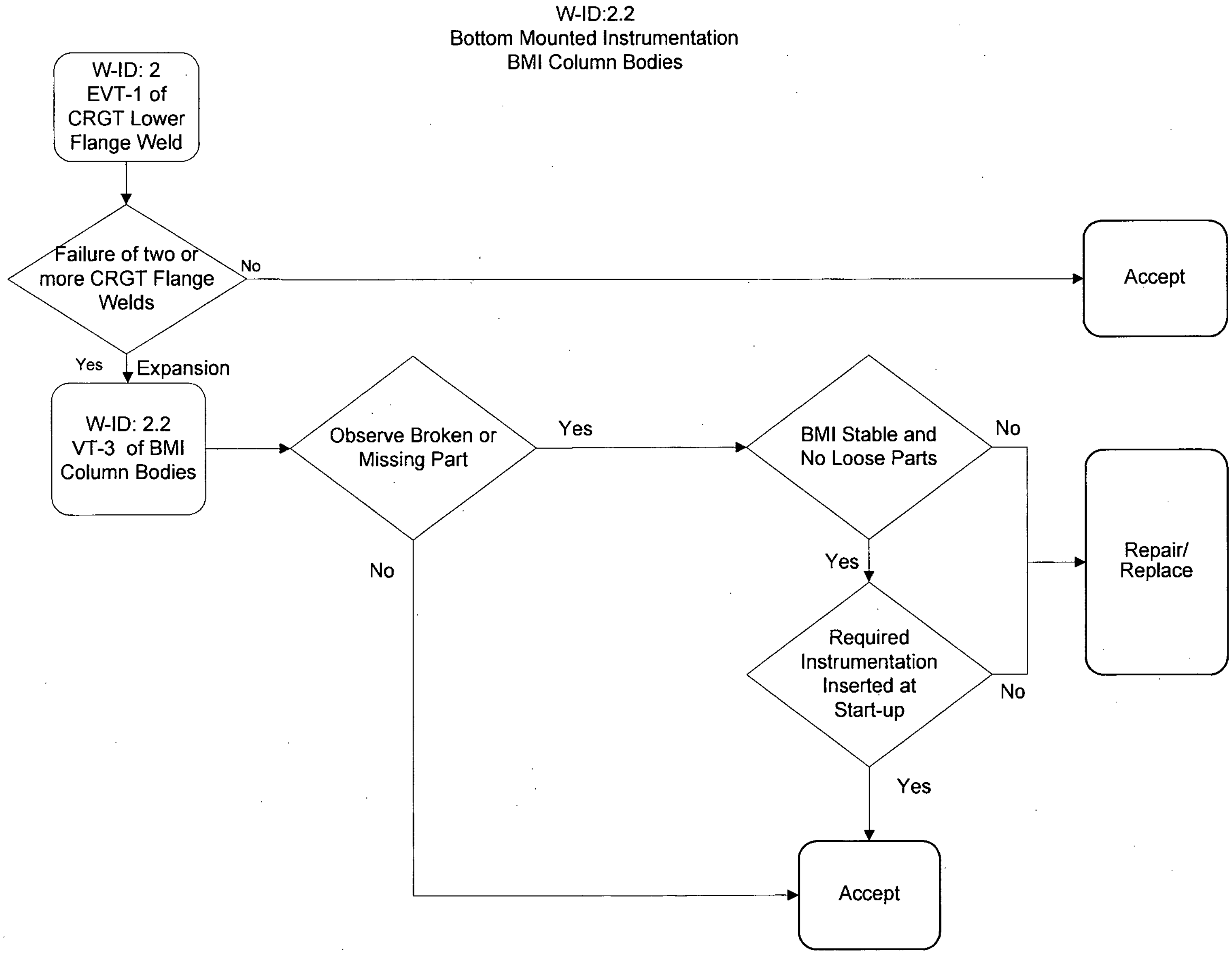
W-ID: 6 Baffle-Former Assembly – Assembly
W-ID: 7 Alignment and Interfacing Components – Internal Hold-down Spring
W-ID: 8 Thermal Sleeve Assembly – Thermal Shield Flexures

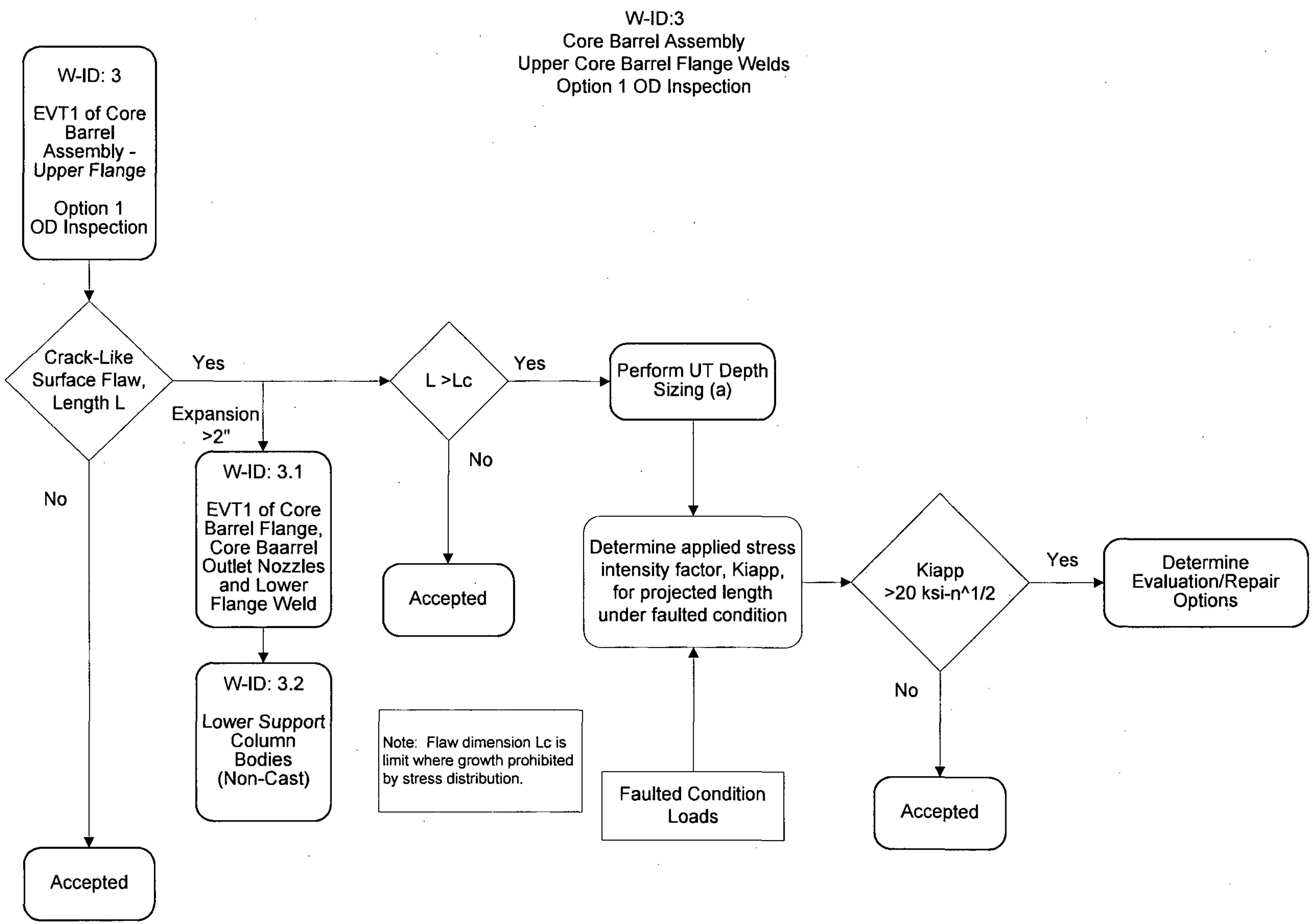
W-ID:1
Control Rod Guide Tube Assembly
Guide Plates (Cards)



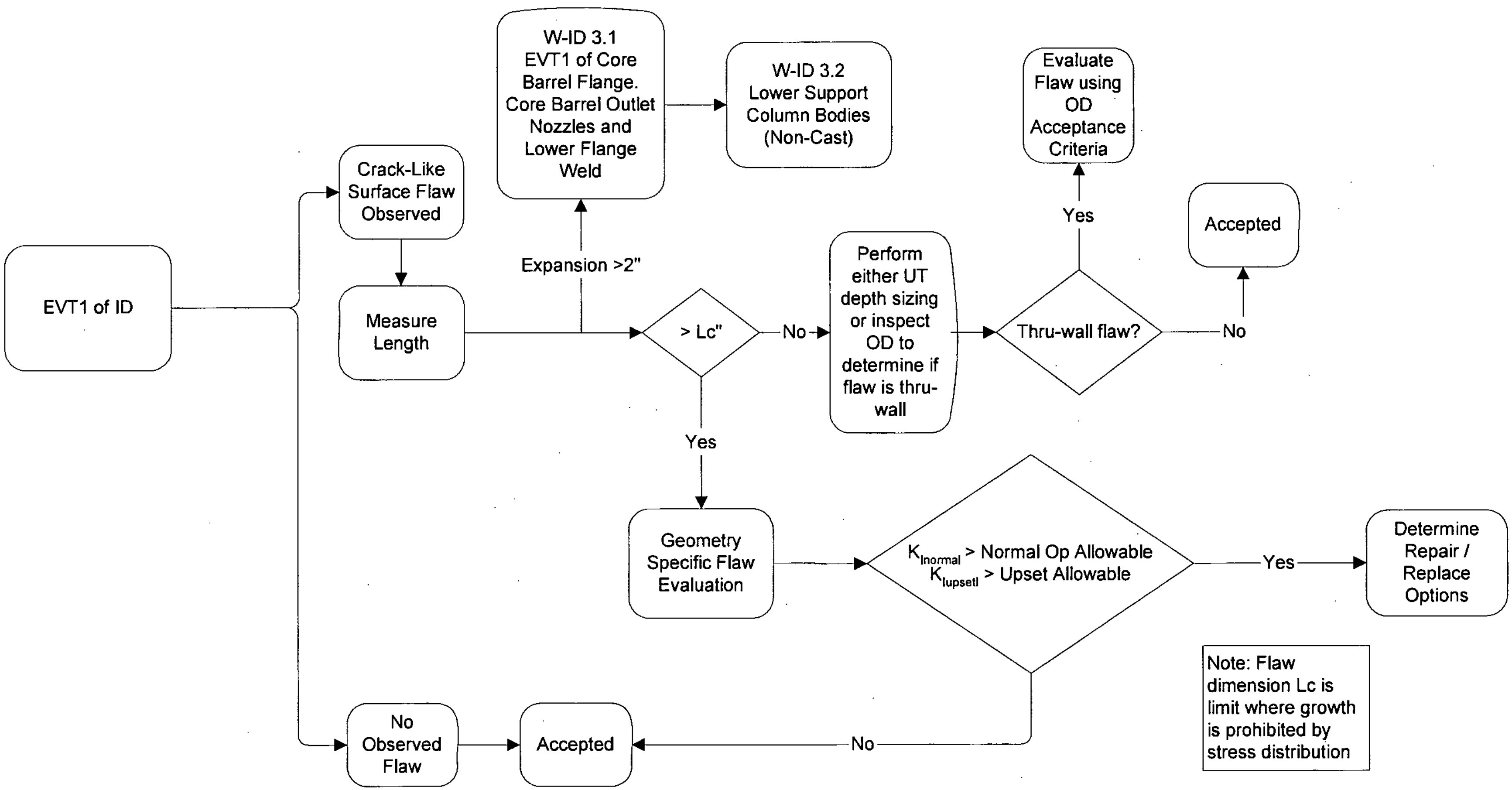

W-ID: 2
Control Rod Guide Tube Assembly Lower Flange Welds
W-ID: 2 EVT-1 of Control Rod Guide Tube Assembly
Lower Flange Weld
Crack-like flaw in one or more support welds
No
Yes
Accept
Yes
Pattern of Unfailed Welds Bounded by Acceptable Pattern
Mitigate
W-ID: 2.1 VT-3 of Cast Lower Support Column Components
W-ID: 2.2 VT-3 of BMI Column Bodies
Table of Acceptable Weld Patterns
Minimum Weld Analysis
Yes
Select Candidate Pattern of Failed Welds
Determine Guide Tube Assembly Deflection with Failed Welds Removed from Analysis
Evaluate: Allowable deflection from Design Analysis
Pattern Acceptable?
No
Table of Rejected Weld Patterns
Select Limiting Faulted Condition

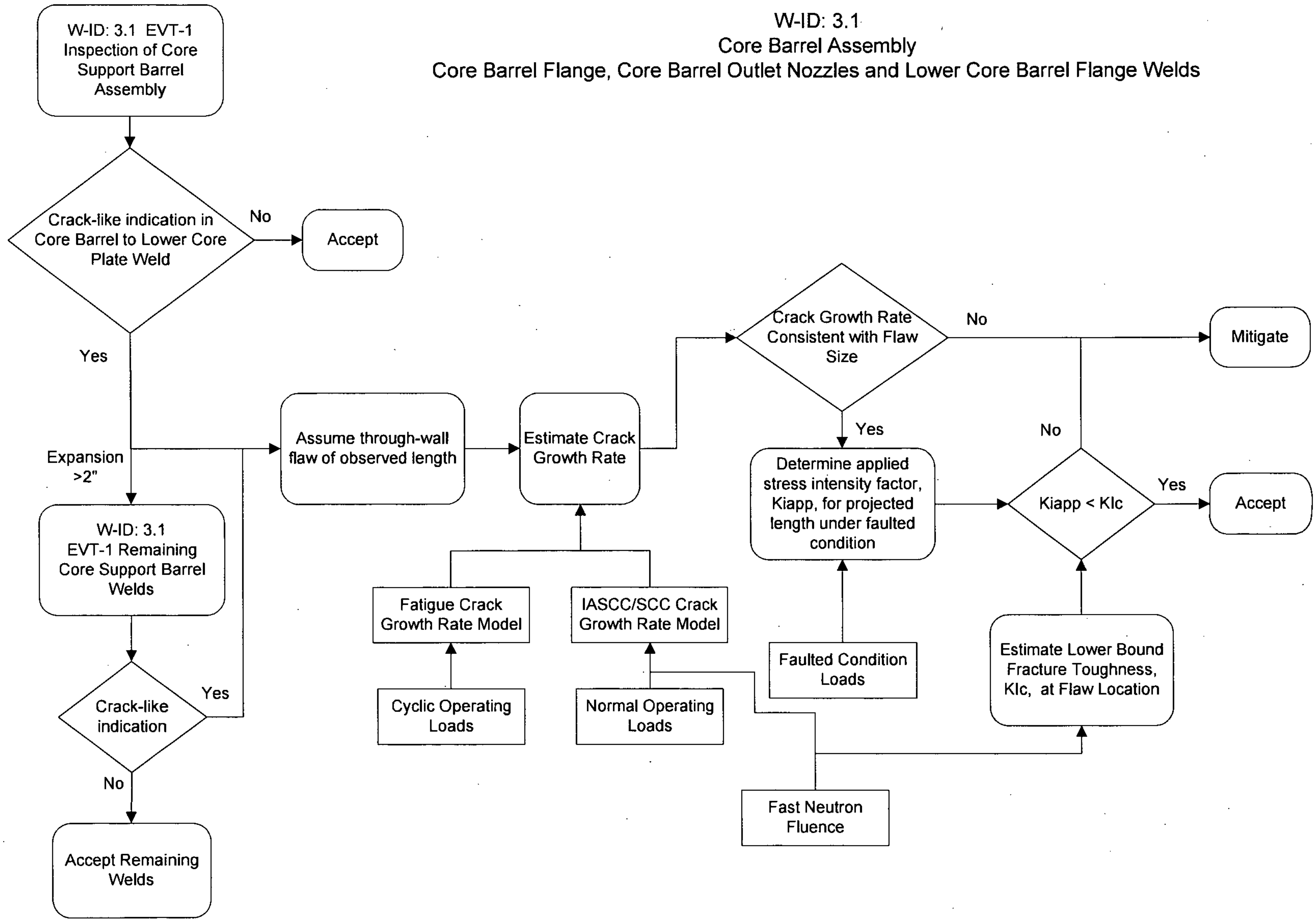

W-ID: 2.1
Lower Support Structure
Lower Support Column Bodies (Cast)
CE-ID: 2
EVT-1 of CRGT
Lower Flange
Weld
Failure of two or more
CRGT Flange Welds
No
Yes
W-ID: 2.1
EVT-1 of Lower
Support Column
Cast Components
Observed Pattern of
Unfailed Columns
Bounded by Acceptable
Pattern
Yes
No
Calculate safety margin:
Margin = N-Nreq-Nf
N = total number of columns
Nreq = columns in minimum pattern
Nf = failed columns
Margin > (N - Nreq)/2
Yes
No
Accept
Mitigate
Table of Acceptable
Column Patterns
Minimum Column Analysis
Select Candidate Column
Pattern
Determine Core
Support Plate
Displacement with
failed Columns
Removed from
Analysis
Evaluate:
Column peak stress
Integrity of Core Support
Pattern
Acceptable?
Yes
No
Table of Rejected
Column Patterns
Select Limiting Faulted
Condition

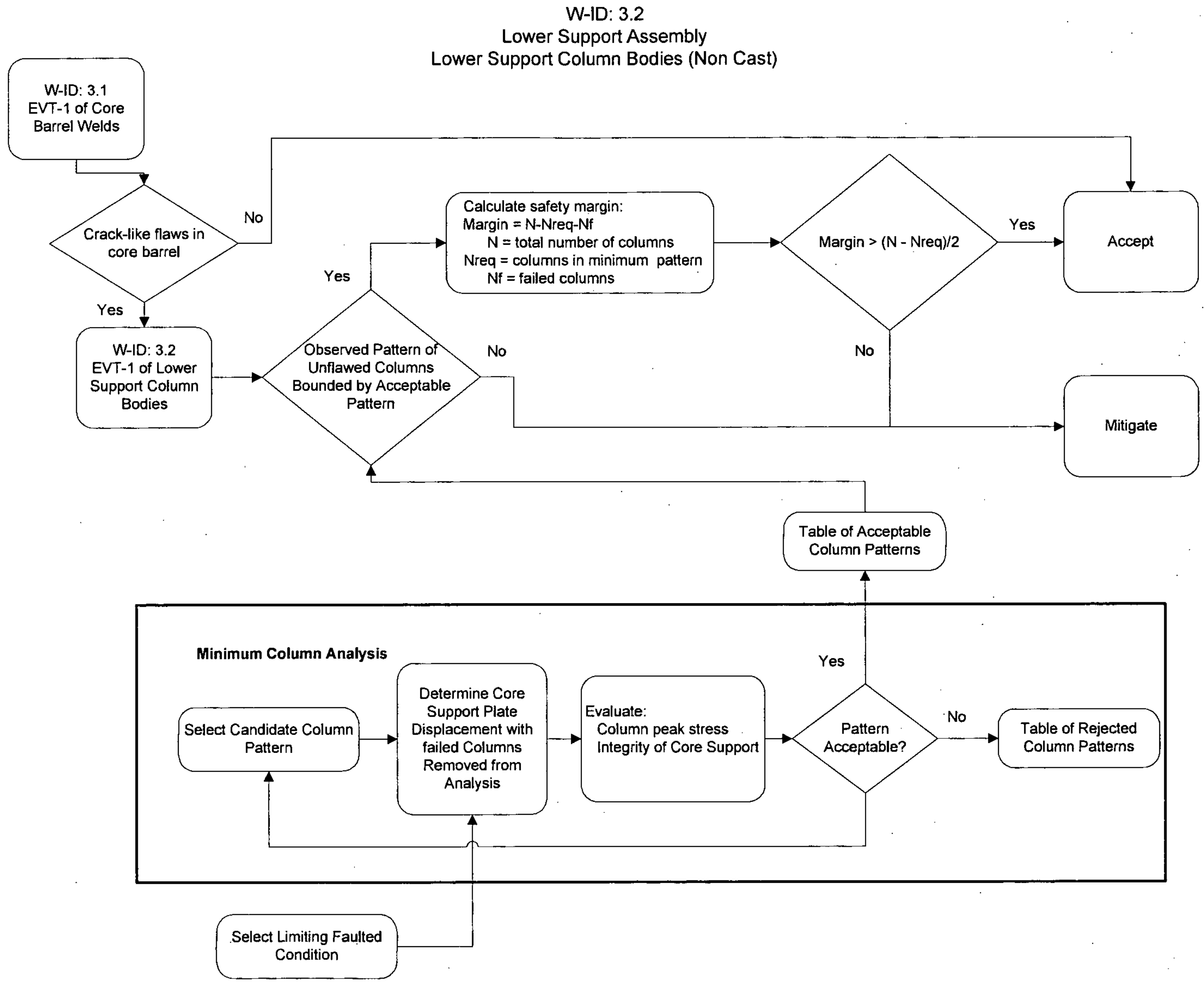

W-ID:2.2
Bottom Mounted Instrumentation
BMI Column Bodies
W-ID: 2
EVT-1 of
CRGT Lower
Flange Weld
Failure of two or more CRGT Flange Welds
No
Accept
Yes
Expansion
W-ID: 2.2
VT-3 of BMI
Column Bodies
Observe Broken or Missing Part
Yes
BMI Stable and No Loose Parts
No
Repair/
Replace
No
Yes
Required Instrumentation Inserted at Start-up
No
Yes
Accept


W-ID:3
Core Barrel Assembly
Upper Core Barrel Flange Welds
Option 1 OD Inspection
W-ID: 3
EVT1 of Core Barrel Assembly - Upper Flange
Option 1 OD Inspection
Crack-Like Surface Flaw, Length L
Yes
Expansion >2"
No
W-ID: 3.1
EVT1 of Core Barrel Flange, Core Baarrel Outlet Nozzles and Lower Flange Weld
W-ID: 3.2
Lower Support Column Bodies (Non-Cast)
Accepted
L >Lc
Yes
No
Accepted
Note: Flaw dimension Lc is limit where growth prohibited by stress distribution.
Perform UT Depth Sizing (a)
Determine applied stress intensity factor, Kiapp, for projected length under faulted condition
Faulted Condition Loads
Kiapp >20 ksi-n^1/2
Yes
No
Determine Evaluation/Repair Options
Accepted

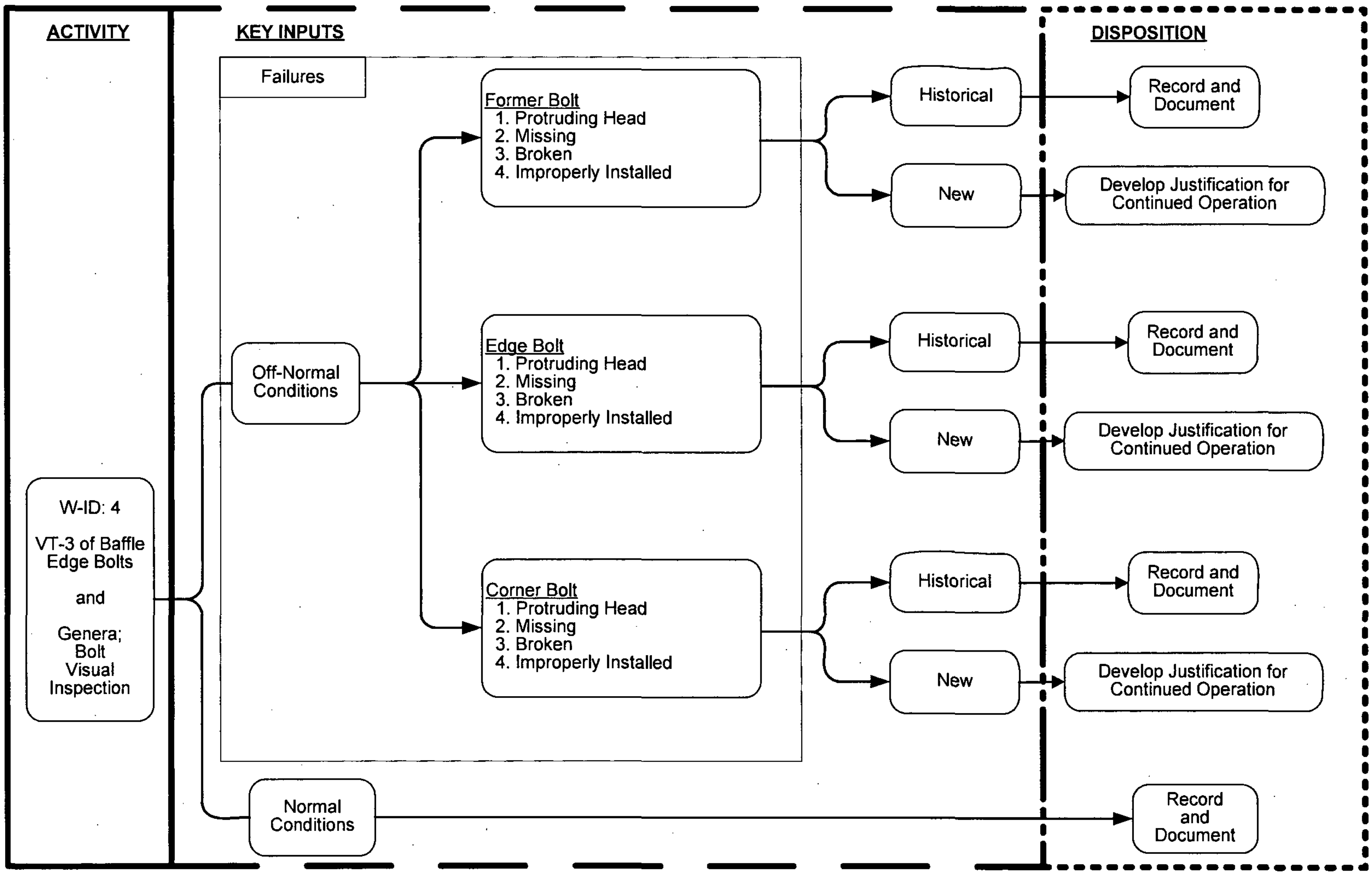
W-ID: 3
Core Barrel Assembly
Upper Core Barrel Flange Weld
Option 2: ID Inspection

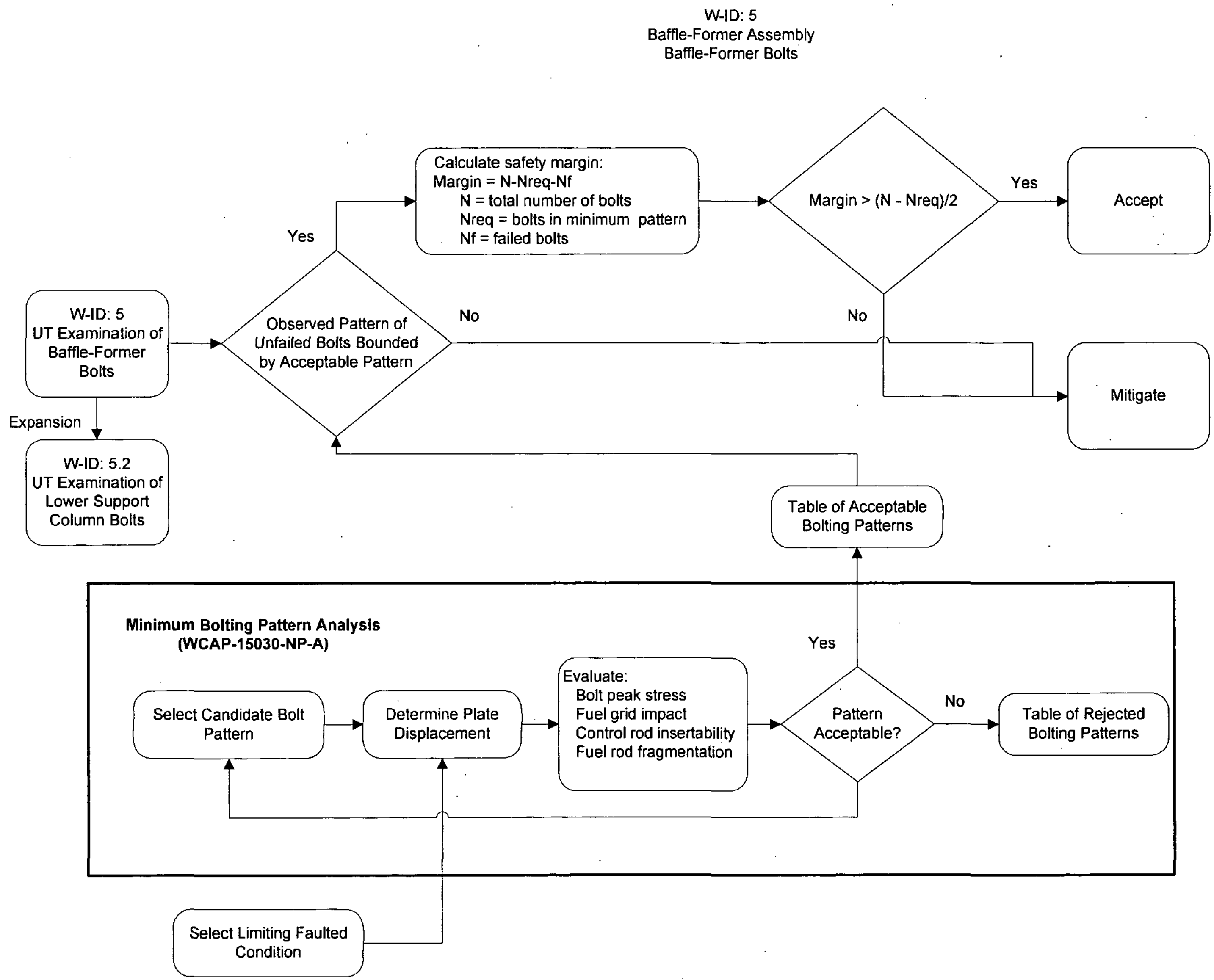



W-ID: 3.1
Core Barrel Assembly
Core Barrel Flange, Core Barrel Outlet Nozzles and Lower Core Barrel Flange Welds
W-ID: 3.1 EVT-1 Inspection of Core Support Barrel Assembly
Crack-like indication in Core Barrel to Lower Core Plate Weld
No
Accept
Yes
Expansion >2"
W-ID: 3.1 EVT-1 Remaining Core Support Barrel Welds
Crack-like indication
Yes
No
Accept Remaining Welds
Assume through-wall flaw of observed length
Estimate Crack Growth Rate
Fatigue Crack Growth Rate Model
IASCC/SCC Crack Growth Rate Model
Cyclic Operating Loads
Normal Operating Loads
Crack Growth Rate Consistent with Flaw Size
No
Mitigate
Yes
Determine applied stress intensity factor, Kiapp, for projected length under faulted condition
Faulted Condition Loads
Fast Neutron Fluence
Kiapp < KIc
No
Yes
Accept
Estimate Lower Bound Fracture Toughness, KIc, at Flaw Location

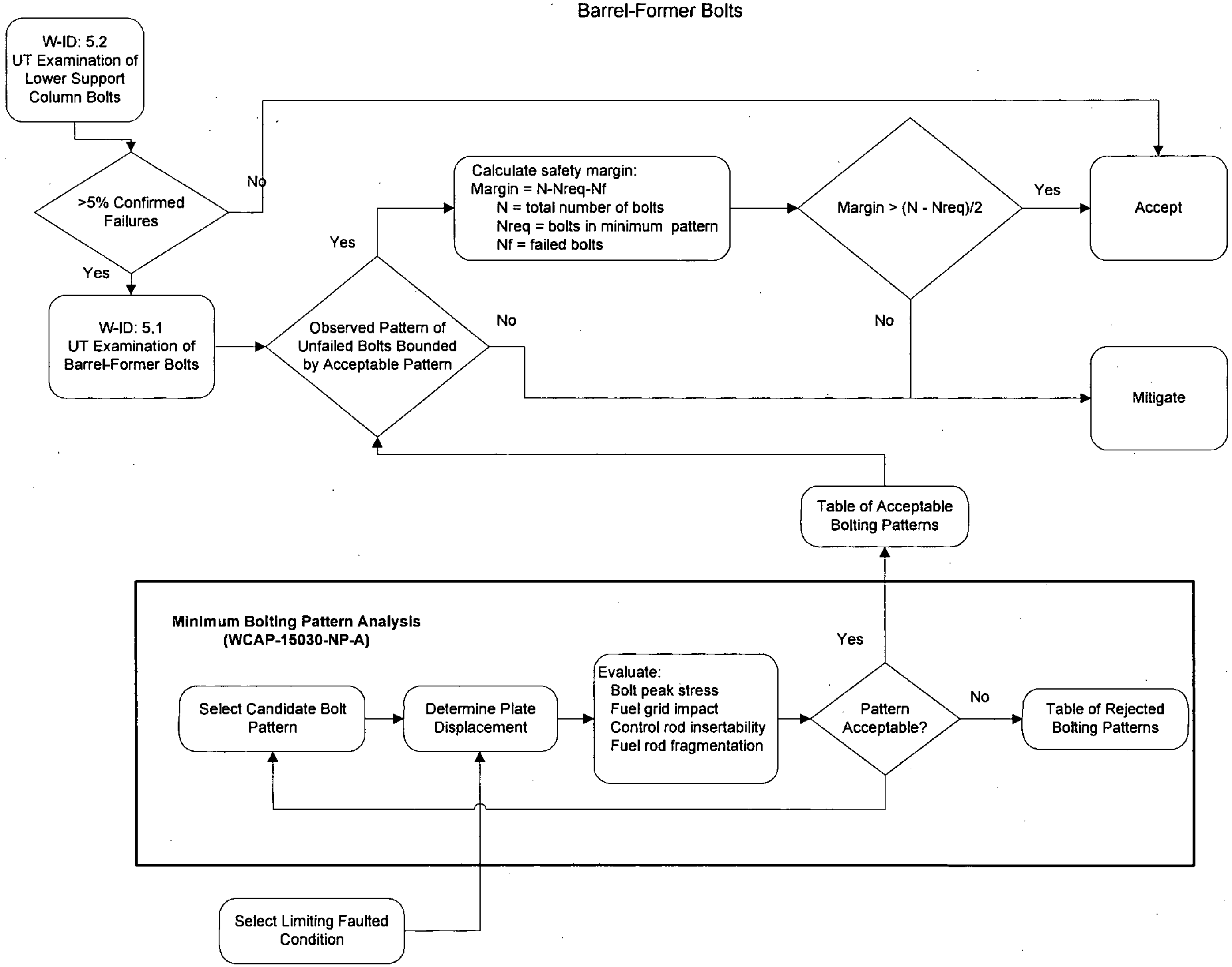

W-ID: 3.2
Lower Support Assembly
Lower Support Column Bodies (Non Cast)
W-ID: 3.1
EVT-1 of Core
Barrel Welds
Crack-like flaws in
core barrel
No
Yes
W-ID: 3.2
EVT-1 of Lower
Support Column
Bodies
Observed Pattern of
Unflawed Columns
Bounded by Acceptable
Pattern
Yes
No
Calculate safety margin:
Margin = N-Nreq-Nf
N = total number of columns
Nreq = columns in minimum pattern
Nf = failed columns
Margin > (N - Nreq)/2
Yes
No
Accept
Mitigate
Table of Acceptable
Column Patterns
Minimum Column Analysis
Select Candidate Column
Pattern
Determine Core
Support Plate
Displacement with
failed Columns
Removed from
Analysis
Evaluate:
Column peak stress
Integrity of Core Support
Pattern
Acceptable?
Yes
No
Table of Rejected
Column Patterns
Select Limiting Faulted
Condition

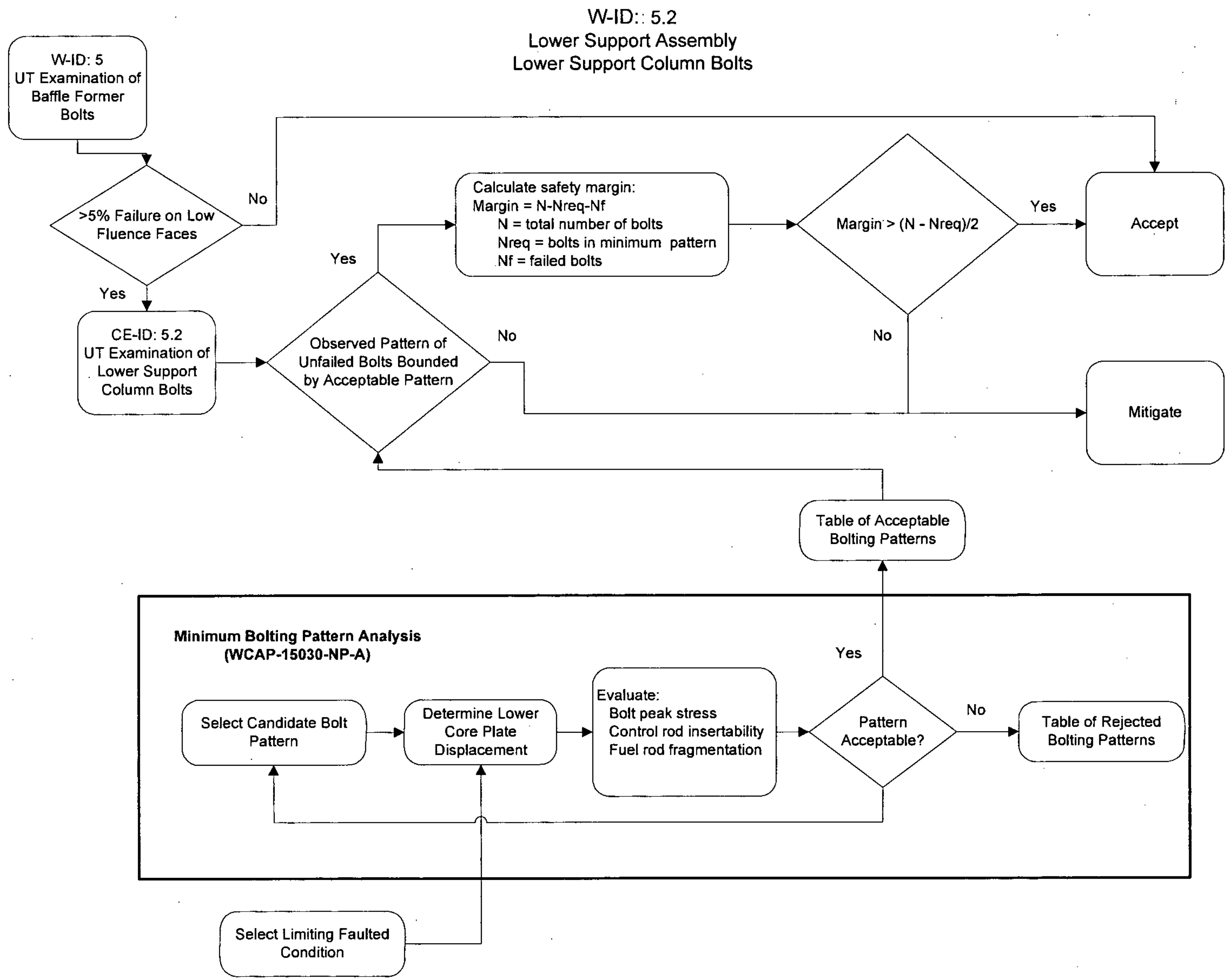
W-ID: 4
Baffle-Former Assembly
Baffle Edge Bolts
FMEA Layout for Bolts



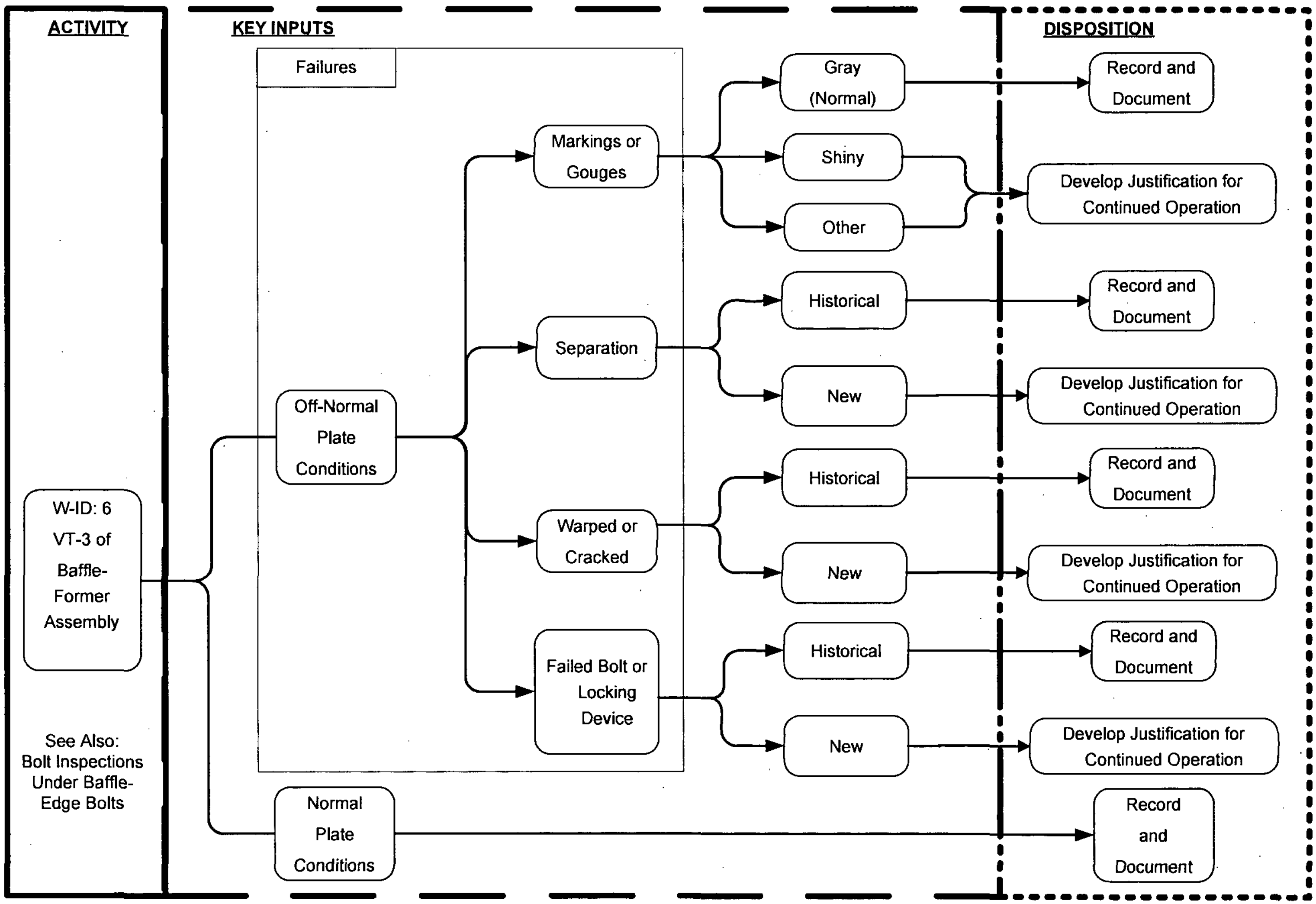

W-ID: 5
Baffle-Former Assembly
Baffle-Former Bolts
Calculate safety margin:
Margin = N-Nreq-Nf
N = total number of bolts
Nreq = bolts in minimum pattern
Nf = failed bolts
Margin > (N - Nreq)/2
Yes
Accept
Yes
W-ID: 5
UT Examination of
Baffle-Former
Bolts
Observed Pattern of
Unfailed Bolts Bounded
by Acceptable Pattern
No
No
Mitigate
Expansion
W-ID: 5.2
UT Examination of
Lower Support
Column Bolts
Table of Acceptable
Bolting Patterns
Minimum Bolting Pattern Analysis
(WCAP-15030-NP-A)
Yes
Select Candidate Bolt
Pattern
Determine Plate
Displacement
Evaluate:
Bolt peak stress
Fuel grid impact
Control rod insertability
Fuel rod fragmentation
Pattern
Acceptable?
No
Table of Rejected
Bolting Patterns
Select Limiting Faulted
Condition

W-ID: 5.1
Baffle-Former Assembly
Barrel-Former Bolts

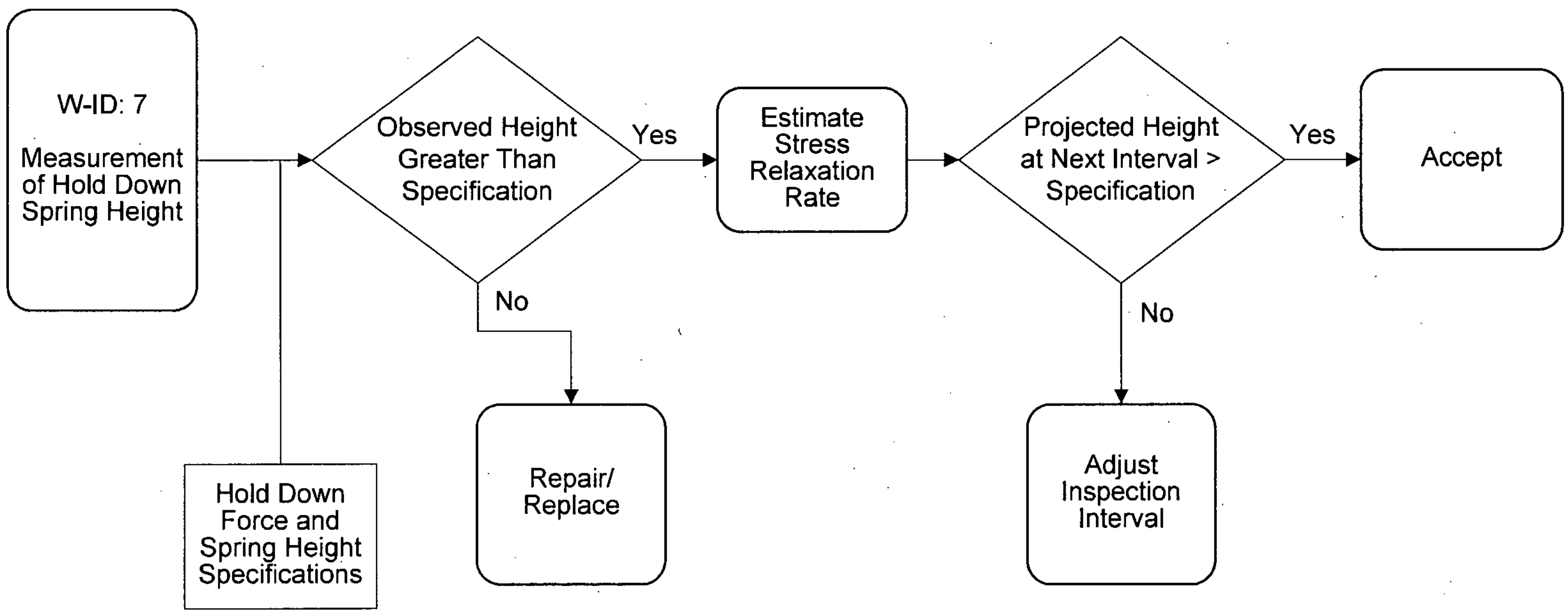



W-ID:: 5.2
Lower Support Assembly
Lower Support Column Bolts
W-ID: 5
UT Examination of Baffle Former Bolts
>5% Failure on Low Fluence Faces
No
Yes
CE-ID: 5.2
UT Examination of Lower Support Column Bolts
Observed Pattern of Unfailed Bolts Bounded by Acceptable Pattern
Yes
No
Calculate safety margin:
Margin = N-Nreq-Nf
N = total number of bolts
Nreq = bolts in minimum pattern
Nf = failed bolts
Margin > (N - Nreq)/2
Yes
No
Accept
Mitigate
Table of Acceptable Bolting Patterns
Minimum Bolting Pattern Analysis
(WCAP-15030-NP-A)
Select Candidate Bolt Pattern
Determine Lower Core Plate Displacement
Evaluate:
Bolt peak stress
Control rod insertability
Fuel rod fragmentation
Pattern Acceptable?
Yes
No
Table of Rejected Bolting Patterns
Select Limiting Faulted Condition

W-ID: 6
Baffle-Former Assembly
FMEA Layout



W-ID: 7
Alignment and Interfacing Devices
Internals Hold Down Spring



W-ID: 8
Thermal Shield Assembly
Thermal Shield Flexures