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Revision 3

August 2016

Westinghouse BWR Control Rod CR 99 Licensing Report – Update to Mechanical Design Limits

Westinghouse

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TABLE OF CONTENTS

LIST OF TABLES v

LIST OF FIGURES vi

EXECUTIVE SUMMARY vii

SUMMARY OF CHANGES viii

1 PURPOSE 1-1
2 INTRODUCTION 2-1
2.1 BASIC WESTINGHOUSE DESIGN 2-1
2.2 LICENSING BACKGROUND 2-1
2.3 CURRENT/FUTURE DEVELOPMENTS 2-2
3 DEFINITIONS 3-1
3.1 CR 99 CONTROL ROD 3-1
3.2 CONFORMANCE METHODS 3-1
3.3 CRITERIA 3-1
3.4 CRITICAL ATTRIBUTES 3-1
3.5 DESIGN REQUIREMENTS 3-1
4 DESIGN REQUIREMENTS 4-1
4.1 GENERAL 4-1
4.2 CONFORMANCE METHODS 4-1
5 MATERIALS EVALUATION 5-1
5.1 CRITICAL ATTRIBUTES 5-1
5.2 CRITICAL ATTRIBUTES DISCUSSION 5-1
5.2.1 Rod Wing and Handle Material 5-1
5.2.2 Button and Roller Material 5-1
5.2.3 Absorbing Materials 5-1
5.2.4 Velocity Limiter 5-2
5.2.5 Coupling Socket 5-2
5.3 MATERIALS CRITERIA AND DISCUSSION 5-3
5.3.1 Materials Criterion 1 (MA-1) 5-3
5.3.2 Materials Criterion 2 (MA-2) 5-3
5.3.3 Materials Criterion 3 (MA-3) 5-4
6 MECHANICAL EVALUATION 6-1
6.1 CRITICAL ATTRIBUTES 6-1
6.2 ATTRIBUTES DISCUSSION 6-1
6.2.1 Blade Thickness 6-1
6.2.2 Hole Diameter 6-1
6.2.3 Hole Pitch 6-2
6.2.4 Hole Depth 6-2
6.2.5 Minimum Outer Wall Thickness 6-2
6.2.6 Hole Ligament Thickness 6-2
6.2.7 Moment of Inertia 6-2
6.2.8 Mass of the Complete Control Rod 6-2
6.2.9 Mass of the Control Rod Without the Velocity Limiter and Socket 6-3

TABLE OF CONTENTS (cont.)

6.2.10 Control Rod Design Temperature 6-3
6.2.11 Control Rod Design Pressure 6-3
6.2.12 Handle Design 6-3
6.2.13 Welds 6-3
6.3 MATERIALS STRENGTH PROPERTIES 6-3
6.4 LINEAR ELASTIC ANALYSIS 6-4
6.4.1 Service Limits 6-4
6.5 LIMIT LOAD ANALYSIS 6-5
6.5.1 Level A Service Limit 6-5
6.5.2 Level B Service Limit 6-5
6.5.3 Level D Service Limit 6-5
6.5.4 Shake Down and Cyclic Response 6-5
6.6 MECHANICAL CRITERIA AND DISCUSSION 6-6
6.6.1 Mechanical Criterion 1 (ME-1) 6-6
6.6.2 Mechanical Criterion 2 (ME-2) 6-7
6.6.3 Mechanical Criterion 3 (ME-3) 6-10
6.6.4 Mechanical Criterion 4 (ME-4) 6-12
6.6.5 Mechanical Criterion 5 (ME-5) 6-13
7 PHYSICS EVALUTION 6-31
7.1 CRITICAL ATTRIBUTES 7-1
7.2 ATTRIBUTES DISCUSSION 7-1
7.2.1 Total Rod Worth 7-1
7.2.2 Shutdown Margin (SDM) 7-2
7.2.3 LPRM Detector Signal Change 7-3
7.2.4 Nuclear End of Life 7-3
7.3 PHYSICS CRITERIA AND DISCUSSION 7-3
7.3.1 Physics Criterion 1 (PH-1) 7-3
7.3.2 Physics Criterion 2 (PH-2) 7-4
7.3.3 Physics Criterion 3 (PH-3) 7-4
7.3.4 Physics Criterion 4 (PH-4) 7-5
8 OPERATIONAL EVALUATION 8-1
8.1 CRITICAL ATTRIBUTES 8-1
8.2 ATTRIBUTES DISCUSSION 8-1
8.2.1 Nominal Wing Thickness 8-1
8.2.2 Maximum Button Thickness 8-1
8.2.3 Maximum Wing Span 8-1
8.2.4 Maximum Velocity Limiter Diameter (With Rollers Installed) 8-1
8.2.5 Total Weight 8-1
8.2.6 Overall Length 8-1
8.2.7 Velocity Limiter/Coupling Design 8-2
8.2.8 Handle Design 8-2
8.2.9 Envelope 8-2

TABLE OF CONTENTS (cont.)

8.3 OPERATIONAL CRITERIA AND DISCUSSION 8-3
8.3.1 Operational Criterion 1 (OP-1) 8-3
8.3.2 Operational Criterion 2 (OP-2) 8-3
8.3.3 Operational Criterion 3 (OP-3) 8-3
8.3.4 Operational Criterion 4 (OP-4) 8-4
8.3.5 Operational Criterion 5 (OP-5) 8-4
8.3.6 Operational Criterion 6 (OP-6) 8-4
8.3.7 Operational Criterion 7 (OP-7) 8-5
8.3.8 Operational Criterion 8 (OP-8) 8-5
9 REFERENCES 9-1

LIST OF TABLES

Table 4-1 Design Requirements/Criteria Matrix 4-2
Table 5-1 Materials Related Critical Attributes for the CR 99 Design 5-4
Table 5-2 Materials Criteria 5-5
Table 6-1 Mechanical Critical Attributes for CR 99 Designs 6-16
Table 6-2 Mechanical Criteria 6-17
Table 7-1 Physical Critical Attributes for CR 99 Designs 7-6
Table 7-2 Physics Criteria 7-7
Table 8-1 Operational Critical Attributes for CR 99 Designs 8-7
Table 8-2 Operational Criteria 8-8

LIST OF FIGURES

Figure 6-1. Sketch Showing Load History (EOL) to be used in Fatigue Assessment 6-11
Figure 6-2. FEM Model of Handle 6-19
Figure 6-3. Helium Release vs ^{10}B-Depletion 6-20
Figure 6-4. Design Pressure Curve 6-21
Figure 6-5. Model of the Blade Wing Structure 6-22
Figure 6-6. Not Used 6-23
Figure 6-7. Seismic Scram Insertion Test, D-Lattice 6-24
Figure 6-8. Seismic Scram Insertion Test, C-Lattice 6-25
Figure 6-9. Seismic Scram Insertion Test, S-Lattice 6-26
Figure 6-10. FE-Model Used to Analyze Seismic Loads on Control Rod 6-27
Figure 6-11. Loads Applied in EOL Seismic Analysis of Control Rod. 6-28
Figure 6-12. Radial ^{10}B-Depletion Profile to Determine Absorber Pin Swelling Loads 6-29
Figure 6-13. Results Showing No Collapse at BOL 6-30
Figure 6-14. Results Showing No Collapse at EOL 6-31
Figure 8-1. Control Rod Tolerance Envelope D-Lattice, Base Design 8-10
Figure 8-2. Control Rod Tolerance Envelope C-Lattice, Base Design 8-11
Figure 8-3. Control Rod Tolerance Envelope S-Lattice, Base Design 8-12

EXECUTIVE SUMMARY

Revision 3 of this topical report incorporates the changes identified during the NRC audit of WCAP-16182, Revision 2, held on May 17-20, 2016. The changes included in Revision 3 of this report are identified in the Summary of Changes section and are marked with rev bars in the right margin.

Optimization of design dimensions together with updated calculations have shown that Westinghouse CR 99 boiling water reactor (BWR) control rods can be operated to significantly higher Mechanical End of Life (MEOL) and Nuclear End of Life (NEOL) than previously approved, with MEOL exceeding NEOL for all service conditions. This report (Revisions 1 through 3) introduces an update to the set of mechanical design requirements previously reviewed and approved in WCAP-16182-P-A, "Westinghouse BWR Control Rod CR 99 Licensing Report," Revision 0 (March 2005). Together, the revised design requirements and criteria form a set of design bases consisting of design requirements, criteria, and verification methods which continue to ensure acceptable performance of the Westinghouse CR 99 BWR control rods.

[

]a,c

Structural capacity of the control rod shall be proven with respect to criteria defined in 2013 ASME Boiler and Pressure Vessel Code (BPVC) Section III Subsection NB-3200. This implies that Westinghouse needs to expand the applicability of analyses that are used for structural verification of the control rod from only linear elastic analysis to linear elastic analysis, limit load analysis, or plastic analysis. All three analyses are defined to be valid for satisfying the structural capacity requirements for a control rod.

The individual changes included in this revision are described and summarized in the following section.

SUMMARY OF CHANGES

Revision 2 to WCAP-16182-P incorporates the following list of changes:

Structural capacity of the control rod shall be proven with respect to criteria defined in 2013 ASME BPVC Section III Subsection NB-3200. This implies that Westinghouse needs to expand the applicability of analyses that are used for structural verification of the control rod from only linear elastic analysis to linear elastic analysis, limit load analysis, or plastic analysis. All three analyses are defined to be valid for satisfying the structural capacity requirements for a control rod. [

]a,c

Tables, in general, have been divided and updated to be consistent with the added design options.

Table 6-1 has been revised due to a change in the helium pressure calculation methodology. Control rod design pressure values have been updated to be consistent with the new methodology. [

]a,c

Figure 6-1 has been added to reflect fatigue assessment.

Figure 6-2 has been revised to reflect a change in software versions, it was formerly Figure 6-1.

Figures 6-3 and 6-4 have been revised to reflect a change in helium pressure calculation methodology, they were formerly Figures 6-2 and 6-3.

Figure 6-5 has been revised to reflect the new finite element model used for calculations of the absorber portion of the control rod, it was formerly 6-4.

Figure 6-6 has been deleted as it pertained to former Equation (6.9), which was also deleted, it was formerly Figure 6-5.

Former Figures 6-6 through 6-8 have been re-numbered from 6-7 through 6-9.

In subsection 7.2.2, changes reflect the new outline of CR 99 absorber zones and associated changes to the shutdown margin calculation involving the weighted sum of the various zones.

[

]a,c

In subsection 7.3.4, changes are provided for clarification and to [

]a,c

In Table 7-1, Physical Critical Attributes for CR 99 Designs, values have been updated to be consistent with updated calculations.

In subsection 8.3.8, details of the 3rd generation CR 99 surveillance program have been included.

[

]a,c

In Section 9 the References 14, 16-22, 24-25, 28-30, 34 are not used. References 39-43 have been added.

Revision 3 to WCAP-16182-P incorporates the following list of changes:

In the Executive Summary, a paragraph is added to explain the purpose of Revision 3 of this topical report. Also, in the first paragraph, expanded the extent of the update to Revision 1 through Revision 3.

In subsection 4.1, information has been added stating that the control rod is not classified as a Class 1 component (also a related change in Section 6).

Table 4-1 has been updated regarding Applicable Criteria on design and manufacturing of the control rod.

In Section 6, explanatory text is added why the control rod is not classified as a Class 1 component.

In subsection 6.5.2 regarding Level B service limit, explanatory text is added.

In subsection 6.6.4.2 reference to References 40 and 41 has been added.

In subsection 6.6.5, discussion has been added describing the results of seismic analyses of the control rod.

Figures 6-10 through 6-14 have been added to include information on the seismic analyses.

In Section 9, the title for Reference 38 has been corrected. Reference 44 has been added.

1 PURPOSE

The purposes of this report are to:

1. Present a set of design requirements for Westinghouse BWR control rods. Given these design requirements, a set of measurable criteria is established which, if met, ensure that the design requirements are met. These design requirements and criteria together form a set of design bases for Westinghouse control rods for use in BWRs.

2. Evaluate the CR 99 design against the measurable criteria to ensure that the design meets the design bases for Westinghouse control rods for BWRs.

2 INTRODUCTION

2.1 BASIC WESTINGHOUSE DESIGN

The basic Westinghouse control rod design for which the Westinghouse experience base is applicable and for which this Licensing Topical Report is intended consists of a control rod which:

1. Has horizontal absorber holes drilled in solid stainless steel wings,

2. Uses guide pads (buttons) or no guide pads rather than the upper pins and rollers used in the Original Equipment Manufacturers (OEM) control rods,

3. [ ]a,c

4. Has a velocity limiter,

5. Weighs less than the design weight for the control rod drive,

6. Has a handle the same as the one it is replacing, or has a core grid support which allows all four surrounding bundles to be removed without needing a blade guide to hold the control rod in place,

7. Has an initial worth within [ ]a,c of the initial worth of the control rod that it is replacing, and

8. Does not negatively impact the ability of the Core Monitoring System to monitor the core (i.e., [ ]a,c).

2.2 LICENSING BACKGROUND

The initial design Westinghouse control rod, designated as CR 70, is described in Reference 1. This design contained only boron carbide (B_4C) as a neutron absorber. Due to the potential for B_4C swelling-induced cracking in the rod tip even when a control rod is fully withdrawn, subsequent designs have contained hafnium (which does not swell when irradiated) in the tips of the rods. The CR 70 design is no longer manufactured. Nevertheless, many of these rods have operated well, and are still in operation, in Swedish built Westinghouse reactors.

Reference 2 describes the next Westinghouse design, CR 82, for use in D-Lattice GE BWRs. This design contains hafnium in the top six inches of the rod, with a total rod worth within 5 percent of the original control rods. With the exception of the hafnium tip, it is essentially the same design as the rod described in Reference 1. Use of this rod design has been approved by the NRC in Reference 3.

Reference 4 discusses the use of the CR 82 design in C-Lattice GE BWRs. This design is similar to the D-Lattice rod design in concept, with differences in geometry and envelope dimensions due to differences in lattice designs. Use of this rod design has been approved by the NRC in Reference 5.

Reference 6 discusses: (1) a design (CR 85) that incorporates hafnium along the outer edge of the rod as well as in the top six inches as used in previous designs, and (2) use of Westinghouse control rods in BWR/6 reactors. NRC approval is documented in Reference 7.

WCAP-16182, Rev. 0 (Reference 38) describes the CR 99 design, which has basically the same features as for instance the CR 82 design mentioned above with the following exceptions:

1. The [ ]^{a,c} as absorber material in the CR 99 design instead of B_4C powder and hafnium rodlets used in the CR 82 design.

2. The use of AISI 316L stainless steel (SS) material in the blade wings of the CR 99 design instead of the AISI 304L SS used in the CR 82 design. This change of material is discussed in Reference 8.

2.3 CURRENT/FUTURE DEVELOPMENTS

Westinghouse's extensive experience with the basic Westinghouse control rod design encompasses more than 40 years in BWR reactors of all vendors. The basic design discussed in the previous section has proven to be an excellent design, and serves as the basis for future designs. Past improvements, as well as foreseeable future improvements, will involve incremental changes on the basic design such that the large experience base of proven designs can be applied to any new design.

Control rod inspections (References 9 through 12) showed an increased potential for CR 82 control rod cracking for rods used in high duty (e.g., Control Cell Core) positions in the core. "High duty" is defined as a location where the control rod is deeply inserted into the core for a significant fraction of the cycle. Rods used in this manner receive high doses of thermal and fast neutrons in a short time when deeply inserted in the core. The fast neutron dose is not measured by current core monitoring systems since it does not lead directly to control rod ^{10}B depletion, but it is well known that fast neutron irradiation makes stainless steel susceptible to irradiation assisted stress corrosion cracking (IASCC).

Thus, an improved design designated CR 99 has been introduced to counteract the potential life shortening IASCC phenomenon. This design uses [ ]^{a,c} as absorber material instead of B_4C powder and hafnium rodlets. AISI 316L SS is the blade wing material. AISI 316L SS has proven to be more resistant to IASCC than AISI 304L SS (Reference 8). This has been shown both in materials experiments and in control rod operation.

3 DEFINITIONS

3.1 CR 99 CONTROL ROD

CR 99 is a control rod design whose critical attributes are presented in Sections 5 through 8 of this report. A large data base of operating experience shows that these rods meet the design requirements, listed in Section 4.1, for Westinghouse control rods in BWRs.

3.2 CONFORMANCE METHODS

These are various methods by which it is possible to verify that the CR 99 design meets specific criteria. These methods include experience, testing, analyses, and inspection.

3.3 CRITERIA

Criteria are a set of quantifiable, measurable standards which, if met, ensure that the design requirements are met.

3.4 CRITICAL ATTRIBUTES

Critical attributes are those attributes (dimensions, materials, design values, etc.) which, if changed, have the potential to affect fit, form, or function of the control rod.

3.5 DESIGN REQUIREMENTS

Design requirements are a set of general guidelines for the design of Westinghouse control rods which, if met, ensure that Westinghouse control rods will operate as required in D-, C-, and S-Lattice BWRs.

4 DESIGN REQUIREMENTS

4.1 GENERAL

The design requirements for Westinghouse BWR control rods to be used in BWRs are:

1. The control rod is compatible with the Control Rod Drive (CRD) system, coupling device, fuel, fuel channels, associated core internals, and rod handling equipment.

2. The control rod is designed such that rod worth and transient operation (e.g., scram and free fall velocity) are consistent with the plant safety analyses.

3. The control rod is designed with mechanical stability and materials such that scram capability is maintained throughout control rod life.

4. The control rod is designed such that currently used tools can monitor core power distribution and burnup.

5. The control rod is designed such that total life cycle dose due to its use (activation product dose, direct dose, and disposal dose) is minimized.

6. The design and manufacture of the control rods are in accordance with applicable codes and standards, including applicable parts of the ASME BPVC. Although the control rod is not classified as a Class 1 component, the structural qualification is based on stress criteria defined in ASME III Subsection NB-3200.

Given the above design requirements, a set of measurable criteria is established which, if met, ensure that the design requirements are met. These criteria are given in Sections 5 through 8. Table 4-1 lists the design requirements along with their related criteria.

These criteria together with the design requirements form a set of design bases for Westinghouse control rods for use in BWRs.

4.2 CONFORMANCE METHODS

Conformance to the acceptance criteria (and ultimately the design requirements) is ensured by at least one of the following methods:

1. Experience with identical or similar design(s)
2. Testing of prototypes, specific features, etc.
3. Analyses
4. Inspection

Of these conformance methods, experience is the preferred approach. The experience approach provides the most applicable, directly comparable method for verification of conformance to criteria. This is why,

in general, design changes are made in small, incremental steps so that the experience base of previous designs remains valid and applicable to new designs.

Where the experience base does not exist or the time to obtain such a base is too long, testing of prototypes as well as specific control rod features may be undertaken. Analyses are used (1) to supplement testing, (2) to extend test results to other product lines or designs, or (3) in lieu of testing when testing is not practical or is prohibitively expensive, and the analytical tools available are known to give credible results.

Inspection is typically used to verify the first three methods rather than directly as a conformance method. Inspection allows for increasing the accuracy of analyses, verifying results of tests, and updating the experience base. Inspections may also lead to improved designs through detection of previously unknown or unanticipated problems that would not have been detected if inspections had not been done.

Table 4-1. Design Requirements/Criteria Matrix

Design Requirement	Applicable Criteria[1]
The control rod is compatible with the CRD system, coupling device, fuel, fuel channels, and rod handling equipment.	MA-2, 3 OP-1, 2, 3, 4
The control rod is designed such that rod worth and transient operation (e.g., scram and free fall velocity) are consistent with the plant safety analyses.	ME-3, 5 PH-1, 2, 3, 4 OP-2, 5, 6
The control rod is designed with mechanical stability and materials choices such that mechanical function is maintained throughout the life of the control rod.	MA-2 ME-1 through 5 OP-7, 8
The control rod is designed such that currently used tools can monitor core power distribution and burn-up.	PH-3, 4
The control rod is designed such that total life cycle dose due to its use (activation product dose, direct dose, and disposal dose) is minimized.	MA-1
The design and manufacture of the control rod fulfill applicable codes and standards, including applicable parts of the ASME BPVC.	ME-1 through 5

Note:

1. Criteria Nomenclature is as follows:
   MA-xx Materials Criteria (See Section 5)
   ME-xx Mechanical Criteria (See Section 6)
   PH-xx Physics Criteria (See Section 7)
   OP-xx Operational Criteria (See Section 8)

5 MATERIALS EVALUATION

5.1 CRITICAL ATTRIBUTES

The critical attributes for materials related items are given in Table 5-1. The materials used in the CR 99 design are also included in the table.

5.2 CRITICAL ATTRIBUTES DISCUSSION

5.2.1 Rod Wing and Handle Material

Use of AISI 316L SS for the rod wing and handle is based on extensive in-reactor experience with the material. Better resistance to IASCC of AISI 316L SS has made it the preferred blade wing material (Reference 8). Since this material is in the reactor and subject to neutron activation, limits on cobalt concentration are set to minimize the release of cobalt to the primary coolant as well as minimize direct doses due to disposal.

5.2.2 Button and Roller Material

These components are subject to contact and are designed to slide or ride against other material. Thus the button and roller material must be wear resistant. Original equipment control rods in GE BWRs were made of material containing high cobalt concentrations (50% to 60%). While acceptable from the wear standpoint, they released unacceptable amounts of cobalt into the reactor coolant. An Electric Power Research Institute (EPRI) project identified a non-cobalt material, Alloy X-750, as an acceptable material for use in fabricating these components. This material has been the material of choice, with the specified limited cobalt content, for the CR 99 control rod. Extensive in-reactor experience, confirmed during post irradiation examinations, has shown this material to perform as required. During the last 20 years, AISI 316L SS has also been used in control rod buttons. Operational experience with this material is also very good.

Operational experience has also demonstrated that the control rods can be operated without a top button. No wear on any component, control rod or fuel channels, has been observed (Reference 13). Additionally, no wear has been observed after Reference 13 was issued.

5.2.3 Absorbing Materials

Extensive in-reactor experience with boron carbide (B_4C) powder has been amassed on Westinghouse BWR control rods. In-pile measurements of helium gas pressure have confirmed the validity and conservatism of the helium release model used in the analyses.

With CR 99, Westinghouse has introduced [ ]^{a,c} This can be compared to the highest density of powder, about 70%, or standard sintering density of about 73%.

In a control rod with B_4C powder, the powder densifies during operation and also swells due to neutron absorption reactions. Westinghouse experience is that the competing effects of powder densification and swelling can result in the swelling powder contacting the surrounding stainless steel, possibly causing IASCC.

[

]a,c

Reference 14 is updated and superseded by Reference 33, which describes the outline of the CR 99 control rod for an S-Lattice BWR6 reactor. CR 99 control rods have accumulated significant operating experience in BWRs.

5.2.4 Velocity Limiter

The design of the velocity limiter is very important to the control rod drop accident analysis. The design of this important component is discussed in Section 8 of this report. From a materials standpoint, the velocity limiter must be made from a material which can be readily cast, machined to final dimensions, and attached to the rod wings. Since it is in contact with primary coolant, cobalt content must also be controlled. The velocity limiter for the CR 99 is manufactured from cast American Iron and Steel Institute (AISI) 304L SS.

Extensive in-reactor experience with all Westinghouse control rods has shown the acceptability of this material for the velocity limiter.

5.2.5 Coupling Socket

The design of the coupling socket is important for proper operation of the control rod. The design of this component is discussed in Section 8 of this report. The coupling socket must be made from a material which can be machined to final dimensions and has sufficient strength to keep the control rod coupled to the drive mechanism. The coupling socket is manufactured from Alloy X-750. Extensive in-reactor experience with this material has shown its acceptability for the coupling socket.

5.3 MATERIALS CRITERIA AND DISCUSSION

The following criteria are shown in Table 5-2 along with the conformance method(s) required to confirm that the criteria are met. CR 99 evaluation results are also provided.

5.3.1 Materials Criterion 1 (MA-1)

Criterion

No material shall be used which results in a larger total rod lifetime dose (direct + indirect) than the material which it is to replace. If it does, compensatory measures must be implemented in some other material(s) to reduce total rod dose to meet this criterion.

Discussion

This criterion ensures that all Westinghouse control rod designs will have at least the same (relative to OEM rods) characteristics with respect to cobalt release during operation, dose received during replacement and preparation for disposal, and disposal-related radiological parameters (dose and curie content).

Evaluating the dose impact for a new material only involves a verification that it contains a lower quantity of radioactive components (e.g., cobalt) than the material that it is replacing. For less obvious material changes, the investigation may require the use of the Westinghouse computer model BKM-CRUD (Reference 15) to determine the impact.

5.3.2 Materials Criterion 2 (MA-2)

Criterion

Rod wing material shall be better than or equal to original blade wing material (Type 304L stainless steel) with respect to stress corrosion cracking, particularly susceptibility to fast neutron IASCC.

Discussion

This criterion and its conformance methods ensure that only materials superior to those already in use are used for rod wings. Thus, it is possible to use past in-reactor experience as a conservative experience base for the new material.

As shown in Table 5-2, the conformance method required to confirm that a material is superior is testing and experience. Previous in-reactor experience with the proposed material and/or testing (e.g., in-pile material tests, autoclave tests, lead control rods, etc.) provides confidence that a material is superior, but the ultimate proof is long term use in its final form in control rods in the reactor. For this reason, the lead control rods containing critical components with new material need to be inspected to confirm results of pre-use testing and adequacy of the experience base.

5.3.3 Materials Criterion 3 (MA-3)

Criterion

Components shall be made of materials compatible with connected and interfacing materials and components.

Discussion

This criterion ensures that the design will be compatible with existing in-reactor materials.

Evaluation to confirm compliance with this criterion will ensure that materials related considerations (e.g., differences in thermal expansion, wear properties, etc.) do not create problems.

Table 5-1. Materials Related Critical Attributes for the CR 99 Design

Materials Critical Attribute	D-, C-, and S-Lattice Material or Value
Rod Wing and Handle Material	AISI 316L SS
Cobalt Content	[ ]a,c
Impurities	[ ]a,c
Velocity Limiter Roller Material	Alloy X-750
Cobalt Content	[ ]a,c
Button Material	Alloy X-750, AISI 316L SS or No Button
Cobalt Content	[ ]a,c
Absorbing Materials	
Boron Carbide	[ ]a,c [ ]a,c Placed in holes drilled in stainless steel
Velocity Limiter	Cast AISI 304L SS
Cobalt Content	[ ]a,c
Coupling Socket	Alloy X-750
Cobalt Content	[ ]a,c

Table 5-2. Materials Criteria

Criterion	Conformance Method(s)[1]	D-, C- and S-Lattice CR 99
(MA-1) No material shall be used which results in a larger total rod lifetime dose (direct + indirect) than the material which it is to replace. If it does, compensatory measures must be implemented in some other material(s) to reduce total rod dose to meet this criterion	Analyses	The materials chosen for CR 99 minimize Co. The two largest contributors to dose are the rollers/buttons (due to movement across other material) and the wings (largest surface). • With respect to the rollers/buttons, the materials chosen (Alloy X-750 and/or AISI 316L SS) have much less Co than the Stellite material in the original rods (see subsection 5.2.2). • With respect to the wing material, the CR 99 has 1/3 of the surface area of the OEM blades. This, combined with a [ ]a,c limit on Co, ensures that this criterion is met for CR 99. Based on the above, the CR 99 rod meets this criterion.
(MA-2) Rod wing material shall better than or equal to original blade wing material (AISI 304L SS) with respect to stress corrosion cracking, particularly susceptibility to fast neutron IASCC.	Experience Testing Inspection	Material testing as well as control rod operating experience have proven AISI 316L SS to be a better material than AISI 304L SS with respect to IASCC (Reference 8). On this basis, the CR 99 rod meets this criterion.
(MA-3) Components shall be made of materials compatible with connected and interfacing materials and components.	Experience Testing Analyses	An extensive experience base has shown that the design meets this criterion, i.e., no problems with latching, normal rod movement, scram (as seen by rod insertion times within Technical Specification limits), or abnormal corrosion. On this basis, the CR 99 rod meets this criterion.

Note:
1. See Section 4.2 for a discussion on Conformance Methods.

6 MECHANICAL EVALUATION

The mechanical evaluation shall verify that the load capacity of the control rod is higher than the design limits. Westinghouse uses stress criteria defined in 2013 ASME BPVC Section III Subsection NB-3200 in the mechanical evaluation of the control rod. The control rod is not classified as a Class 1 component because:

1. The control rod does not form a part of the reactor coolant pressure boundary.
2. The control rod is designed to be a spare part.
3. Small cracks in the control rod blade will not lead to catastrophic failure of the control rod or jeopardize its safety function.

The safety margin exhibited by the control rod, that is inherent in the application of the stress criteria defined in ASME BPVC Section III, NB-3200, is considered reasonable for its design. Therefore, the mechanical evaluation of the control rods is performed by demonstrating that the stress criteria according to 2013 ASME BPVC Section III NB-3200 are satisfied. This implies that linear elastic stress analysis, limit load analysis, or plastic analysis can be used in the verification.

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6.1 CRITICAL ATTRIBUTES

The critical attributes for mechanical related items are shown in Table 6-1. The attribute values for CR 99 are also included.

6.2 ATTRIBUTES DISCUSSION

6.2.1 Blade Thickness

The blade thickness has an impact on the load capacity of the control rod. Therefore, the value of the blade thickness in the load capacity evaluation shall be chosen in accordance with [ ]a,c of the load capacity.

6.2.2 Hole Diameter

The hole diameter affects the wall thickness as well as the ligament thickness between adjacent holes, which has an impact on the load capacity of the control rod. Therefore, the value of the diameter in the load capacity evaluation shall be chosen in accordance with [ ]a,c of the load capacity.

Furthermore, the hole diameter affects the internal volume of the control rod blade, which influences the internal helium pressure. Therefore, the internal helium pressure calculation shall be [ ]a,c

[

]a,c

6.2.3 Hole Pitch

This parameter affects the ligament thickness between holes, which has an influence on load capacity. Therefore, the ligament thickness in the load capacity evaluation shall be chosen in accordance with [ ]a,c of the load capacity.

6.2.4 Hole Depth

The hole depth is the primary parameter used by Westinghouse to control the scram load capacity of a control rod. Varying the hole depth can change the control rod load capacity of two otherwise identical control rods. Therefore, the value of the hole depth in the load capacity evaluation shall be chosen in accordance with [ ]a,c of the load capacity.

Furthermore, the hole depth affects the internal volume of the control rod blade, which influences the internal helium pressure. [

]a,c

6.2.5 Minimum Outer Wall Thickness

The thickness of the outer wall has impact on the internal helium pressure capacity. Therefore, the minimum value of the outer wall thickness in the load capacity evaluation shall be chosen in accordance with [ ]a,c of the load capacity.

6.2.6 Hole Ligament Thickness

The ligament thickness between holes has impact on the load capacity. Therefore, the value of the ligament thickness in the load capacity evaluation shall be chosen in accordance with [ ]a,c of the load capacity.

6.2.7 Moment of Inertia

The moment of inertia is important mainly with respect to seismic behavior and the ability of the control rod to be inserted in the core during a seismic event. Westinghouse has performed scram insertion tests into oscillating cores, which prove that the insertion time criteria are fulfilled. Therefore, the moment of inertia of any new control rod design shall be lower than or equal to the moment of inertia of the scram tested control rods. As such, the scram capability of any new control rod design is bounded by performed scram insertion tests.

6.2.8 Mass of the Complete Control Rod

This parameter is important in determining axial stresses in the control rod during a scram event. Maximum control rod mass shall be used in the determination of scram forces.

6.2.9 Mass of the Control Rod Without the Velocity Limiter and Socket

This parameter is important in determining axial stresses in the control rod during a scram event. Maximum control rod mass, without the velocity limiter and socket, shall be used in the determination of scram forces.

6.2.10 Control Rod Design Temperature

The control rod temperature is set by the design temperature of the plant reactor coolant system. This value is far below any value that could substantially degrade (melt) the material in the control rod.

6.2.11 Control Rod Design Pressure

The control rod design pressure is set by the design of the core (axial depletion profile of the control rod) and plant reactor coolant system. The control rod design pressure is used in the calculation of the stresses across the hole walls due to differential pressure.

6.2.12 Handle Design

Westinghouse has manufactured control rods with both single and double handles. The safety function of the control rods does not depend on the handle design. However, the designs must be: (1) checked for compatibility with the rod handling equipment, and (2) evaluated to ensure that the handle will be able to take the stresses due to normal loading and handling. Note that item (1) is addressed in Section 8, Operational Evaluation.

In general, the original control rods for D-Lattice plants were built with single handles, C-Lattice plants have a mix of single and double handle control rods, and S-Lattice plants have double handle control rods.

6.2.13 Welds

The weld quality factor and fatigue strength reduction factor according to 2013 ASME BPVC Section III Table NG-3352-1 shall apply in the structural verification of welds.

6.3 MATERIALS STRENGTH PROPERTIES

The material strength properties to be used in the stress (load capacity) evaluation shall be at least [ ][a,c] If tensile test data allowing the statistical evaluation of material strength properties does not exist, strength data from Westinghouse material specifications or 2013 ASME BPVC Section II Part D shall be used.

Mechanical stress/strain data for material in irradiated conditions is provided in References 40, 41, and 43.

6.4 LINEAR ELASTIC ANALYSIS

Load capacity evaluation based on linear elastic stress analysis shall be based on 2013 ASME BPVC Section III NB-3200. Primary membrane stress (P_m), local membrane stress (P_L), primary bending stress (P_b) and secondary stress (Q) are defined in 2013 ASME BPVC Section III NB-3213.

The design stress limit, S_m, is given by 2013 ASME BPVC Section II Part D Appendix 2 Table 2-100 (a):

$$S_m = \min\left\{\frac{2 \cdot R_{p0.2}(20^0 C)}{3};\quad 0.9 \cdot R_{p0.2}(T^0 C);\quad \frac{R_m(20^0 C)}{3};\quad \frac{R_m(T^0 C)}{3}\right\} \qquad \text{(Eq 6-1)}$$

where,

$R_{p0.2}$ - yield strength
R_m - ultimate tensile strength
T - denotes property at actual temperature

6.4.1 Service Limits

The stress limits to be applied in the load capacity verification are given in table below. These stress limits are defined in 2013 ASME BPVC Section III NB-3200.

	Level A Service Limit	Level B Service Limit	Level D Service Limit
P_m	S_m	$1.1 \cdot S_m$	S_m*
P_L	$1.5 \cdot S_m$	$1.1 \cdot 1.5 \cdot S_m$	$1.5 \cdot S_m$*
P_L+P_b	$1.5 \cdot S_m$	$1.1 \cdot 1.5 \cdot S_m$	$1.5 \cdot S_m$*
P_L+P_b+Q	$3 \cdot S_m$	$3 \cdot S_m$	-
Reference	NB-3222	NB-3223	Appendix F

Note:
$S_m^* = \min\{2.4 \cdot S_m ; 0.7 \cdot R_m\}$

6.5 LIMIT LOAD ANALYSIS

The limit load analysis is a plastic analysis that uses the definition of an ideal plastic material. According to the lower bound theorem, 2013 ASME BPVC Section III NB-3213.29, a collapse load is defined as any system of stresses that satisfy equilibrium. In practice, this implies that any force converge solution in a plastic finite element analysis that is based on the definition of an ideal plastic material, is a conservative estimation of the lower bound collapse load.

6.5.1 Level A Service Limit

Load capacity for Level A service limit shall be based on a yield strength equal $1.5 \cdot S_m$, 2013 ASME BPVC Section III NB-3228.1. The allowed load is defined to 2/3 of calculated lower bound collapse load.

6.5.2 Level B Service Limit

Load capacity for Level B service limit shall be based on a yield strength equal $1.5 \cdot S_m$, 2013 ASME BPVC Section III NB-3228.1. The allowed load shall be calculated as 1.1x2/3 of calculated lower bound collapse load. The use of the 1.1 factor is the Westinghouse interpretation of the increase in allowable stress intensity defined in 2013 ASME BPVC Section III NB-3223, and is applied for the allowable load capacity.

6.5.3 Level D Service Limit

Load capacity for Level D service limit shall be based on yield strength equal;

$$R_{p0.2} = \min\{2.3 \cdot Sm, 0.7 \cdot R_m\} \qquad \text{(Eq 6-2)}$$

Allowed load shall be calculated as 90% of calculated lower bound collapse load, 2013 ASME BPVC Section III Appendix F-1341.3.

6.5.4 Shake Down and Cyclic Response

Shake down analysis shall be performed based on 2013 ASME BPVC Section III NB-3228.4. The design of the control rod shall be considered acceptable if shake down occurs.

The cyclic response to be used in fatigue assessment shall be calculated as the numerically maximum principal total strain range multiplied by one-half of the modulus of elasticity of the material at the mean value of the temperature of the cycle, 2013 ASME BPVC Section III NB-3228.4 (c).

The Westinghouse interpretation of the fatigue assessment is that it shall be based on the cyclic response. If shake down occurs during the initial load cycles, these load cycles shall be neglected in the fatigue assessment and the stable cyclic response of the structure will be used instead. If stable cyclic response does not evolve, each load cycle shall be evaluated and the linear damage summation shall be used to calculate the cumulative damage due to the cyclic response.

6.6 MECHANICAL CRITERIA AND DISCUSSION

Mechanical design criteria to be met are stress limits (load capacity limits) and fatigue limits for conditions under which the control rod will be operating during its lifetime. Criteria to be used in the stress (load capacity) evaluation are defined in Section 6.4 (linear elastic stress analysis) or in Section 6.5 (limit load analysis).

The loads that the control rods will be subjected to during their lifetime originate from transport and handling, pressure differences between the inside and the outside of the control rod blade wing, scram loads, and loads due to interaction with fuel channels during sever core vibrations. Mechanical Criterion 1 in subsection 6.6.1 through Mechanical Criterion 5 in subsection 6.6.5 specify the loads that a control rod shall be able to withstand without exceeding the limits identified in 6.4.1. Subsection 6.6.3.2 specifies the load history to be used in fatigue assessment of the control rods, the methods used to perform the fatigue assessment, and the design criteria to be satisfied.

6.6.1 Mechanical Criterion 1 (ME-1)

Criterion

Stresses (or loads) on the control rod handle due to normal loading and handling shall not exceed the allowable values for Level A Service Condition any time during the life of the control rod.

Discussion

This criterion ensures that a control rod can be safely moved during receipt, initial installation, shuffling, removal, and preparation for disposal.

In the Westinghouse design, the support and the handle have been integrated with the control rod wings, which means that there is only one vertical weld where the two control rod wings are joined in the lifting handle.

During normal handling operations, the lifting handle is loaded with the weight of the control rod in air. In stress analysis (load capacity analysis), this load is conservatively chosen as a concentrated force at the weld location on the horizontal portion of the handle. The force to be applied on the handle is:

$$F_{lift} = 2 \cdot M_{CR} \cdot g \qquad \text{(Eq 6-3)}$$

where,

2 – is the dynamic lifting factor
M_{CR} – is the mass of the control rod
g – is the standard gravity.

6.6.2 Mechanical Criterion 2 (ME-2)

Criterion

Stresses (or loads) in the control rod wings due to pressure differential shall not exceed the allowable values for the Level A Service Condition anytime during the life of the control rod.

Discussion

This criterion ensures that allowable stress limits (load limits) are met with the maximum outside to inside ΔP at the beginning of life and maximum inside to outside ΔP at the end of life, and at all differential pressures in between that a Westinghouse control rod may be subjected to during its lifetime.

6.6.2.1 Pressure Difference Determination

During reactor operation the gas pressure in the control rod blades increases with ^{10}B depletion. From the initial filling gap pressure the differential pressure, ΔP, gradually increases to its maximum design pressure across the wall of the blades at Mechanical End of Life (MEOL). The differential pressure for which the blade stresses must be calculated is also a function of reactor temperature and system pressure.

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The gas model used is van der Waals model, which is a modification of the ideal gas law:

$$\left(P + a\frac{n^2}{V_{free}^2}\right)\cdot\left(V_{free} - n\cdot b\right) = n\cdot R\cdot T \qquad \text{(Eq 6-4)}$$

where,

the constants take the values a= $3.46\cdot10^{-3}$ Pa m^6/mol^2 and b=$23.71\cdot10^{-6}$ m^3/mol (Reference 39) and

P Internal gas pressure, MPa
V_{free} Void volume, m^3
n Moles of helium released during neutron absorption
R Universal gas constant, 8.314 J/mol K
T Temperature, K

The helium pressure calculation methodology includes a statistical treatment of fractional helium release and irradiation induced solid absorber pin growth, where a 95/95 confidence limit shall be applied for the helium release and absorber pin growth.

The moles of helium released as a function of ^{10}B depletion are calculated using the equation below, which describes the relation between the amount of helium produced and the fraction of helium released:

$$f_{He} = N\left(\max(C_1, C_2 + C_3\cdot U), \sigma\right) \qquad \text{(Eq 6-5)}$$

where,

U is the ^{10}B depletion.

The following parameter values for the fractional helium release model are given in Reference 39.

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[ ] a,c

[

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6.6.3 Mechanical Criterion 3 (ME-3)

Criterion

Stresses (or loads) in the control rod due to scram loads, assuming normal operation of the scram equipment, shall not exceed allowable values defined in subsection 6.4.1 (or section 6.5.1) for Level A Service Condition anytime in life. Stresses (or loads) in the control rod due to scram loads, assuming a failed buffer, shall not exceed allowable values defined in subsection 6.4.1 (or subsection 6.5.2) for Level B Service Condition anytime in life.

Furthermore, fatigue damage in a control rod shall not evolve during the designed lifetime. The lifetime of a control rod is defined at 200 start/stop cycles of the reactor and each start/stop cycle shall include one scram event at cold conditions assuming a failed buffer, and two scram events at warm conditions assuming normal operation of the buffer.

Discussion

These criteria ensure that allowable stress limits (load limits) are met with any plant specific scram load definition throughout the lifetime of any Westinghouse control rod design.

6.6.3.1 Scram Load Definition

During a reactor scram, the control rods are hydraulically inserted into the reactor core and hydro-dynamically slowed down at the end of the stroke. Therefore, a scram load cycle is defined as a compressive scram force (acceleration) followed by a tensile scram force (deceleration). The maximum axial force in the control rod occurs during the deceleration phase at cold reactor conditions, and assuming a failed buffer. A scram during cold conditions (cold scram) is modeled at a temperature of 85°C and at 300°C in case of a scram during normal reactor operation (warm scram).

6.6.3.2 Cyclic Load Combination

This criterion ensures that a control rod will not be damaged by fatigue during its lifetime, which is defined by the NEOL. [

]a,c

a,c

Figure 6-1. Sketch Showing Load History (EOL) to be used in Fatigue Assessment

Fatigue assessment shall be based on fresh material fatigue data according to 2013 ASME BPVC Section II Part D. Linear summation of the calculated damage shall be used to prove that the sum, in the equation below, is always less than 1.0:

$$D = \sum_i \frac{N_i}{N_{f,i}} < 1 \qquad \text{(Eq 6-11)}$$

where,

N_i is the number of cycles and $N_{f,i}$ is the corresponding number of cycles causing fatigue damage according to 2013 ASME BPVC Section II Part D.

6.6.4 Mechanical Criterion 4 (ME-4)

Criterion

Swelling of the B_4C pins should not cause cracking in the blades.

Discussion

This criterion is set as a design goal.

A consequence of neutron absorption is that the absorber pins swell due to the reaction:

$$^{10}B + n \rightarrow {}^{7}Li + {}^{4}He + 2.8\ MeV$$

[

]a,c

6.6.4.1 Pin Swelling Loads

Absorber pins swell as a consequence of the neutron flux, therefore, maximum pin swelling load occurs at the position of maximum fluence and at NEOL.

The load acting on the control rod at the position of maximum fluence originates from a combination of helium pressure, scram load, and absorber pin swelling. Pin swelling shall be included in the finite element model as a volumetric swelling of the absorber pin and its properties shall be based on the models documented in Reference 36.

Primary stresses (or loads) in the control rod blades shall fulfill the criteria for the Level A Service Limit (identified in Section 6.4.1) for a normal scram, including the effects of loads due to pin swelling and helium pressure. The long term behavior of the control rod blade shall be evaluated based on the IASCC prediction model that is under development, as described in section 6.6.4.

[

]a,c

The structural verification includes the irradiation hardening of the control rod blade material at NEOL.

6.6.4.2 Pin Swelling Load Combined with Scram Force and Helium Pressure

Pin swelling load, combined with scram force, and helium pressure has been analyzed. These analyses show that the CR 99 design has sufficient load capacity to withstand such combined loads, per References 40 and 41. Mechanical properties of the absorber pins have been determined by the theoretical modelling shown in Reference 42.

6.6.5 Mechanical Criterion 5 (ME-5)

Criterion

Insertion of the control rod into an oscillating reactor core during a dynamic event shall not influence the control rod worth nor its future operation. The future operation depends on the frequency classification of the dynamic event (Anticipated Operational Occurrence or Accident).

Insertion time shall be less than minimum specified insertion time for the reactor type of consideration.

Discussion

This criterion ensures that the control rods have the capability to shut down the reactor core within the specified time frame and that the reactor core remains in the shutdown condition. The criterion also means that the neutron absorbing B_4C must remain in the design position during control rod insertion and when the control rod is positioned in the core.

The flexibility of the control rod is described by the moment of inertia (MOI). Lower MOI means a more flexible control rod that is easier to insert into the core. Furthermore, a lower MOI also decreases contact forces (frictional forces) between the control rod and its neighboring channels. Westinghouse uses a test rig to simulate control rods being inserted into an oscillating core in order to measure insertion times for specific core designs (lattice type). The insertion time requirement for any control rod design with an equal or lower MOI is verified by tests. Acceptable insertion times for the Westinghouse CR 85 control rod designs for BWR2/3/4, BWR 5, and BWR 6 corresponding to the D-, C-, and S-Lattice [

]a,c are verified in Toshiba laboratory tests under simulated earthquake conditions. The tests verified the insertion times for the Westinghouse CR 85 for vibration amplitudes up to [

]a,c Test results are shown in Reference 23. Since the MOI for the CR 99 is equal or less than that for the CR 85, see Table 6-1, the CR 99 design can be considered bound by the tests on the CR 85.

The impact of seismic events up to [ ]^{a,c} fuel channel deflection has been studied for both D-/S-lattice, as well as the C-lattice CR 99. Cyclic plastic analyses and fatigue assessments show that no cracking will be initiated due to seismic events up to this magnitude. However, pin swelling loads have not been included in these analyses for two reasons. Firstly, at BOL the diameter of the B_4C pins is smaller than the diameter of the absorber holes and, therefore, no contact forces are expected to evolve during core oscillations. Secondly, by EOL the solid B_4C pins would have been transformed into powder, which would compress during the first core oscillation and result in an insignificant swelling force.

In the last few years, channel distortion has been an increasing challenge for BWR plants. The causes of channel distortion include the fast neutron flux gradient across the channel (both axially and radially) and the proximity of a CRBs near the channel. Even with channel distortion, a CRB can be inserted in the core during a scram because the forces driving the blade into the core are significantly higher than the friction forces between the channel and CRB. However, since these friction forces can lengthen the time it takes for the scram to be completed, it can affect the assumptions in the accident analysis. For that reason, the plant closely monitors for the presence of channel distortion. Calculational analyses and periodic scram tests are performed to ensure that no excessive force is needed to insert the CRBs into the core. If the force needed to insert a CRB exceeds a certain conservative level during a scram test, then that blade is completely inserted in the core and declared inoperable for the rest of the cycle.

Stress verification of the control rod when scrammed into an oscillating core has been done by plastic analysis with acceptance criteria according to ASME III Appendix F (Reference 44). Assumptions for the loads applied in the analyses are:

- Constant control rod temperature of 300 °C.
- Loads from absorber pin swelling shall be included based on section 6.6.4.1.
  - BOL, zero swelling forces
  - EOL, maximum quarter segment ^{10}B depletion shall be used to determine the loads from absorber pin swelling
- Helium pressure
  - BOL, helium pressure $P_{helium} = \frac{(300+273)}{(20+273)} \cdot P_{fill}$ where P_{fill} is helium fill pressure after manufacturing.
  - EOL, helium pressure EOL shall be applied
- Maximum core oscillation of [ ]^{a,c} shall be assumed in the analysis
- Scram force shall be included in the analyses assuming normal operation of buffer.

The forces are linearly ramped up as a function of analysis time until collapse of the structure occurs or until sufficient safety margin can be demonstrated. Zero load is assumed at analysis time zero. Loads increase as illustrated in the figures described below.

Figure 6-10 shows the finite element (FE) model used in the analysis of the S- & D-Lattice designs and the deformation caused by the core oscillation. Figures 6-11 and 6-12 shows loads applied in the EOL analysis of the S- & D-Lattice designs.

The model includes a high resolution of details close to the most highly stressed section in the control rod. At other locations the model is simplified with less resolution. BOL analysis is based on fresh material data for the stainless steel and the EOL analysis is based on irradiated material data.

The results presented in Figures 6-13 and 6-14 show that collapse of the CR occurs at loads higher than 1.5 times the loadings specified in Reference 44. The control rod is therefore verified for the specified loads with a margin of at least 50%.

Structural verification of the S- & D-Lattice designs, Reference 40, compared with structural verification of the C-Lattice design shows that the utilization of the C-Lattice design is lower for scram loads, helium pressure loads and absorber pin swelling loads. Furthermore, the geometry of the C-Lattice design is more beneficial with respect to loads originating from absorber pin swelling (smaller diameter of absorber pins and the same wall thickness). Therefore, the seismic analyses of the S- & D-Lattice designs bound the C- Lattice design.

Table 6-1. Mechanical Critical Attributes for CR 99 Designs

a,c

Mechanical Critical Attribute			
Hole diameter			
Hole pitch			
Hole depth			
Minimum outer wall thickness			
Hole ligament thickness			
Non-irradiated HIP pin maximum diameter (d_p)			
Moment of inertia			
Mass of complete control rod (m)			
Mass of control rod without the velocity limiter and socket (m_1)			
Control rod design temperature			
Control rod design pressure (P_y, reactor operation)			
Control rod design pressure, P_i ($\Delta P = P_i - P_y$ for Stress Calculations)			
Handle design			

Table 6-2. Mechanical Criteria

Criterion	Conformance Method(s)	D-Lattice Reference 40	C-Lattice Reference 41	S-Lattice Reference 40

a,c

Table 6–2. Mechanical Criteria (cont.)

a,c

a,c

Figure 6-2. FEM Model of Handle

a,c

Figure 6-3. Helium Release vs ^{10}B-Depletion

a,c

Figure 6-4. Design Pressure Curve

a,c

Figure 6-5. Model of the Blade Wing Structure

Figure not used

Figure 6-6. Not Used

a,b,c

Figure 6-7. Seismic Scram Insertion Test, D-Lattice

a,b,c

Figure 6-8. Seismic Scram Insertion Test, C-Lattice

a,b,c

Figure 6-9. Seismic Scram Insertion Test, S-Lattice

a,c

Figure 6-10. FE-Model Used to Analyze Seismic Loads on Control Rod

a,c

Figure 6-11. Loads Applied in EOL Seismic Analysis of Control Rod.

a,c

Figure 6-12. Radial ^{10}B-Depletion Profile to Determine Absorber Pin Swelling Loads

a,c

Figure 6-13. Results Showing No Collapse at BOL

a,c

Figure 6-14. Results Showing No Collapse at EOL

7 PHYSICS EVALUTION

7.1 CRITICAL ATTRIBUTES

The critical attributes for physics related items are given in Table 7-1. The values for the CR 99 control rod are also included in the table.

7.2 ATTRIBUTES DISCUSSION

7.2.1 Total Rod Worth

Rod worth calculations have been typically done using the PHOENIX code (Reference 26) to allow comparison of Westinghouse control rod worth to the worth of the rod it is replacing at various conditions simulating a range of reactor conditions. Results of these calculations are then used to confirm nuclear compatibility with the core.

PHOENIX single bundle calculations are made at three different conditions simulating various shutdown conditions:

1. Cold, clean critical – corresponding to the limiting shutdown condition,
2. Hot-Full power, zero void – corresponding to a location near the core inlet, and
3. Hot-Full power, 50% void – corresponding to the top of the core.

[

]a,c

For multiple absorber control rods, the calculations are done for each different absorber zone separately. The total control rod worth difference between the Westinghouse control rod and the replaced rod is then a weighted sum of the various zones. The weighting factors describe the axial power distributions and depend on the type of control rod and on the shutdown conditions, cold clean or hot.

The differences between Westinghouse control rods and the replaced rod using the above procedure vary only slightly for any lattice type control rod design as a function of fuel burn-up and fuel type.

7.2.2 Shutdown Margin (SDM)

In general, shutdown margin follows rod worth, i.e., higher worth translates to more shutdown margin. Westinghouse experience has shown the following to be a good estimate of the impact rod worth has on shutdown margin at limiting cold conditions:

[ ]a,c (Eq 7-1)

where,

Δ SDM is the change in SDM, relative to an OEM rod
Δ k_{COLD} is the PHOENIX single bundle cold clean rod worth of the OEM rod

$$\Delta k = \frac{k_{\infty}^{\text{with CR}} - k_{\infty}^{\text{without CR}}}{k_{\infty}^{\text{without CR}}}(\%) \qquad \text{(Eq 7-2)}$$

and RWD is the relative rod worth difference between the Westinghouse control rod and the rod it is replacing

$$\text{RWD} = \frac{\Delta k(\text{West}) - \Delta k(\text{OEM})}{\Delta k(\text{OEM})} \cdot 100(\%) \qquad \text{(Eq 7-3)}$$

[ ]a,c

For multiple absorber control rods, total SDM is a weighted sum of the various zones. [ ]a,c For an example of a CR 99 absorber material outline, see Reference 33. The total SDM change would be (Reference 35):

[ ]a,c

As with the calculation of total rod worth, there is only a slight ΔSDM dependence on fuel burn-up and fuel type.

7.2.3 LPRM Detector Signal Change

This calculation, which indicates the power distribution effect relative to the replaced rod, is also done using the PHOENIX code. Results of this calculation are used to ensure nuclear compatibility and negligible effect on the core monitoring system.

7.2.4 Nuclear End of Life

Many of the reload analyses performed, and core monitoring codes used in plants, assume that all control rods are new, full strength OEM control rods. For this assumption to remain valid for replacement rods, differences in replacement rod initial worth and allowable depletion relative to the OEM rods must be limited. Replacement rod initial worth of 95% to 105% of OEM initial worth, and allowable control rod depletion of 10% loss in reactivity from initial OEM rod worth, have been the historical limits for GE BWRs. Calculation of Westinghouse BWR control rod worth reduction is done using the PHOENIX/XYBDRY method described in Reference 27.

An important mechanical design characteristic of the CR 99 is [

]^{a,c}

References 28-30, updated and supplemented by new References 35-37, show calculated NEOLs for Westinghouse BWR CR 99 control rods based on the defined limit of 10% loss in reactivity from initial OEM rods.

7.3 PHYSICS CRITERIA AND DISCUSSION

The following criteria are shown in Table 7-2 along with the conformance method(s) required to confirm that the criteria are met. CR 99 evaluation results are also shown.

7.3.1 Physics Criterion 1 (PH-1)

Criterion

Total Westinghouse control rod initial worth shall be within [ ]^{a,c} of the initial worth of the control rod it is replacing.

Discussion

This criterion helps ensure that any Westinghouse control rod design has nuclear compatibility with other rods in the core as well as helping to ensure that calculations performed by the installed core monitoring system remain valid. In addition, this criterion ensures that in-reactor response of the rod will be indistinguishable from the rod it replaces.

Results of calculations done for a specific lattice type control rod design vary only slightly as a function of burn-up and fuel type. Thus, calculations done at the time of initial design of a Westinghouse control rod for installation in a representative core will remain valid for the life of the rod and are valid for other similar lattice type cores.

7.3.2 Physics Criterion 2 (PH-2)

Criterion

The effect on shutdown margin due to the use of a Westinghouse control rod shall be such that:

$$SDM_{Westinghouse} \geq [\quad]^{a,c}\ SDM_{Replaced} \qquad \text{(Eq 7-5)}$$

Discussion

This criterion helps ensure that core monitoring and reload related calculations, which are done assuming an OEM control rod is installed, remain valid.

As discussed in subsection 7.2.2, results of calculations done for a specific lattice type control rod design vary only slightly as a function of burn-up and fuel type.

7.3.3 Physics Criterion 3 (PH-3)

Criterion

The difference seen by an LPRM detector due to the use of a Westinghouse control rod relative to the use of the replaced rod in the same location shall be less than or equal to [ ]^{a,c}.

Discussion

This criterion helps ensure that the calculations done by the core monitoring system remain valid as well as ensuring that local power distribution uncertainties are not significantly increased.

7.3.4 Physics Criterion 4 (PH-4)

Criterion

The NEOL for a Westinghouse control rod is reached when its rod worth in any quarter segment decreases to 90% of the initial worth of an OEM control rod in the quarter segment.

Discussion

This criterion helps ensure that core monitoring and reload related calculations which are done assuming a fresh, OEM control rod is installed, remain valid. A value of 90% of initial worth of an OEM rod in any quarter segment has been historically used for this limit in GE BWRs.

Use of a Westinghouse control rod past this historical limit is acceptable as long as the control rod worth is explicitly monitored in appropriate reload and core monitoring codes, mechanical limits for the projected longer life are investigated, and appropriate inspections are carried out after the Westinghouse control rod exceeds the 10% reactivity loss threshold. For such use, end of life for the Westinghouse control rod would occur when either of the following occurs:

- The worth of the rod decreases to the point where fuel costs are negatively impacted (i.e., loading pattern cannot be optimized due to the decreased worth of the rod), or
- A visual inspection detects a crack.

[

]a,c

Table 7-1. Physical Critical Attributes for CR 99 Designs

Physical Critical Attribute	D–Lattice CR 99 Value or Range	C–Lattice CR 99 Value or Range	S–Lattice CR 99 Value or Range
Total rod worth relative to replaced rod			
Shutdown margin relative to replaced rod			
LPRM detector signal change relative to replaced rod			
NEOL, (10% worth decrease from OEM value)			
Top quarter segment			
2nd and 3rd quarter segments			
Bottom quarter segment			

a,c

Table 7-2. Physics Criteria

Criterion	Conformance Method(s)[1]	CR 99, D-, C- and S-Lattice Valuation Results
(PH-1) Total Westinghouse control rod initial worth shall be within [ ]^{a,c} of the initial worth of the control rod it is replacing.	Analyses	See Table 7.1 (meets Criterion)
(PH-2) The effect on shutdown margin due to the use of a Westinghouse control rod shall be such that: $SDM_{Westinghouse} \geq$ [ ]^{a,c} $SDM_{Replaced}$	Analyses	See Table 7.1 (meets Criterion)
(PH-3) The difference seen by an LPRM detector due to the use of a Westinghouse control rod relative to the use of the replaced rod in the same location shall be less than or equal to [ ]^{a,c}.	Analyses	See Table 7.1 (meets Criterion)
(PH-4) The NEOL for a Westinghouse control rod is reached when its rod worth in any quarter segment decreases to 90% of the initial worth of an OEM rod quarter segment.	Analyses	See Table 7.1 (meets Criterion)

Note:

1. See Section 4.2 for a discussion on Conformance Methods.

8 OPERATIONAL EVALUATION

8.1 CRITICAL ATTRIBUTES

The critical attributes for operational related items are given in Table 8-1. The attribute values used for the CR 99 are also included in the table.

8.2 ATTRIBUTES DISCUSSION

8.2.1 Nominal Wing Thickness

The most important dimensional parameter with respect to compatibility with fuel and fuel channels is the control rod envelope discussed in subsection 8.2.9 below. However, nominal wing thickness is also an important parameter that should be examined for different rod designs.

8.2.2 Maximum Button Thickness

Along with the envelope dimensions, this parameter is important with respect to fuel and channel compatibility. The button is the feature which touches the adjacent fuel channels, helping to keep the control rod centered in the gap between the fuel assemblies.

The CR 99 control rod can also be delivered with no button (Reference 13).

8.2.3 Maximum Wing Span

Maximum wing span is important to compatibility of the rod with core internals and CRD components (e.g., fit through the fuel support piece and fit in the guide tube).

8.2.4 Maximum Velocity Limiter Diameter (With Rollers Installed)

This parameter is important in ensuring compatibility with the CRD system, in particular the guide tube. The rollers on the end of the velocity limiter ride against the inside of the guide tube. The maximum diameter of the velocity limiter with the rollers installed must be such that the rod can travel freely up and down in the guide tube without binding.

8.2.5 Total Weight

Total weight for a control rod must be less than that for which the CRD system was designed.

8.2.6 Overall Length

Overall length is important with respect to interfacing with the CRD system and core internals.

8.2.7 Velocity Limiter/Coupling Design

The design of the velocity limiter is important with respect to the free fall velocity assumed in the Control Rod Drop Accident.

Coupling (socket) design is important since this component provides the control rod interface with the CRD system.

8.2.8 Handle Design

Westinghouse has manufactured control rods with both single and double handles. To ensure compatibility with the rod handling equipment, the handle design of the Westinghouse control rod should be checked against the design of the replaced rod.

In general, the original rods for D-Lattice plants were built with single handles, C-Lattice plants have a mix of single and double handle rods, and S-Lattice plants have double handle rods.The control rods can also be delivered with a core grid support, which allows all four surrounding bundles to be removed without needing a blade guide to hold the control rod in place, provided that the control rod is fully inserted. This means that the handle will be extended up to 2.8 in. (72 mm). When the rod is completely inserted, the support will extend into the core grid. When the rod is completely withdrawn, the handle will experience additional neutron fluence compared with the standard handle. This additional fluence does not limit the use of the rod since the handle is not stressed during operation.

8.2.9 Envelope

The envelope figure for a Westinghouse control rod shows the maximum thickness of the blade as well as the maximum allowed twist and bow along the full length of the control rod.

This envelope is checked for every control rod along its full length in a full length test fixture as part of the manufacturing process.

This envelope is important in determining proper rod interface with fuel, fuel channels, and other core internals.

8.3 OPERATIONAL CRITERIA AND DISCUSSION

The following criteria are shown in Table 8-2 along with the conformance method(s) required to confirm that the criteria are met. CR 99 evaluation results are also shown.

8.3.1 Operational Criterion 1 (OP-1)

Criterion

The Westinghouse control rod socket shall be compatible with the existing CRD coupling device (spud).

Discussion

A good coupling design ensures that (1) the control rod can be coupled to the drive when initially installed, (2) the control rod will remain coupled during operation, and (3) the control rod can be uncoupled when the rod is to be shuffled or removed.

8.3.2 Operational Criterion 2 (OP-2)

Criterion

The Westinghouse control rod weight shall be similar to the nominal weight of the OEM rod.

Discussion

The control rod cannot significantly exceed the nominal weight of the OEM rod due to considerations of scram capability, scram times and free fall (rod drop) characteristics. However the control rod shall not be significantly below the weight of the OEM rod due to settling capability, which depends on the weight of the control rod to cause it to settle into its final position during normal insertion and withdrawal.

8.3.3 Operational Criterion 3 (OP-3)

Criterion

The Westinghouse control rod shall be compatible with existing fuel, fuel channels, and core internals.

Discussion

This criterion is important to ensure that normal operation and scram capability are not impacted, i.e., the control rod will not damage surrounding fuel channels, and will fit in the core.

8.3.4 Operational Criterion 4 (OP-4)

Criterion

The Westinghouse control rod shall be compatible with control rod handling equipment.

Discussion

This criterion would only be of concern in cases where the Westinghouse control rod handle design is different from that which it is replacing. Examples would be providing a double handled rod for a plant originally supplied with single handled rods or supplying rods with extended handles.

Compatibility with rod handling equipment is not a safety issue but, nevertheless, must be investigated to ensure that the handling equipment can move, install, and remove the control rods.

8.3.5 Operational Criterion 5 (OP-5)

Criterion

The Westinghouse control rod free fall velocity shall be consistent with the design basis velocity.

Discussion

This criterion (along with OP-2) ensures that any Westinghouse control rod design is consistent with the control rod free fall assumptions in the plant's safety analysis for the Control Rod Drop Accident.

The velocity limiter design for the CR 99 is identical to the design of the OEM control rods.
This, in combination with control rod weights less than those assumed in the design of the CRD system, ensures that the CR 99 meets Criterion OP-5.

In addition, free fall velocity tests of Westinghouse control rods have been performed (Reference 32) that show that Westinghouse control rods meet this criterion.

8.3.6 Operational Criterion 6 (OP-6)

Criterion

The Westinghouse control rod shall not adversely affect scram times and settling capability in the reactor.

Discussion

In conjunction with OP-2, this criterion ensures that scram times will be consistent with those assumed in the plant's safety analyses. In addition, it ensures that any Westinghouse control rod design also settles normally when withdrawn or inserted which, while not a direct safety concern, is a necessary operational consideration.

8.3.7 Operational Criterion 7 (OP-7)

Criterion

Flow-induced vibration of the Westinghouse control rods shall not cause detrimental fretting of the rod or fuel channels.

Discussion

The criterion ensures that control rod vibration, which may be induced by coolant flow in guide tubes and/or in the core, does not have any adverse effect on the control rod or on adjacent fuel channels.

The Westinghouse control rod is designed to have similar clearances to guide tubes and fuel channels as the original control rod. As a result, flow velocities and flow patterns, and thus, rod vibrations will not be significantly changed. In addition, interfacing surfaces between the control rod and channel are designed to have sufficiently large contact area to avoid fretting.

8.3.8 Operational Criterion 8 (OP-8)

Criterion

Mechanical End of Life for all new Westinghouse control rod designs should be greater than or equal to the Nuclear End of Life.

Discussion

This criterion is set as a design goal. Nevertheless, historical in-reactor experience has shown that there is a possibility of unexpected cracking due to B_4C swelling, material cold work, IASCC, etc. In reality, a crack in a Westinghouse control rod has no impact on the safety function of the rod. Rather, the concern is with eventual wash-out of boron carbide, resulting in unmonitored control rod worth reduction. Hot cell examinations and neutron radiography in reactor pools have shown that the loss of B_4C in Westinghouse control rods with B_4C powder (e.g., CR 70) through leaching and washout is very limited in adjacent not cracked holes during the course of one or even several operating cycles. [

]^{a,c}

Westinghouse has a policy to follow lead control rods of each design to high burnups by performing inspections. From these inspections, guidelines for operation and the need for further inspections of the various designs are formulated.

3rd Generation CR 99 Surveillance Program

Inspections have been performed on six leading 3rd generation CR 99 control rods, that were operated in two Swedish BWRs to almost 80% of their nuclear life, with positive results, i.e., no cracking. Following its policy, Westinghouse will continue to inspect the leading rods at higher exposures, close to NEOL, in order to verify the CR 99 design.

An inspection plan has been developed for D, C, and S lattice plants as follows:

- A minimum of two (2) 3rd generation CR 99 control rods will be followed at operation in high duty locations in a D, C, and S-lattice, US or international BWR.

- Additional 3rd generation CR 99 control rods are operated in other US BWRs with the intent that they remain at lower depletion than the two lead-depletion 3rd generation CR 99 control rods at the designated BWRs. Should other control rods at a domestic or international BWRs become the highest depletion in the BWR fleet, they will become the control rods inspected per this surveillance program.

- The two lead-depletion control rods will be irradiated, achieving as close to NEOL as practical (target minimum 90% of end of life).

- For refueling outages in which the depletion of the lead 3rd generation CR 99 control rods are greater than 75% of design life, the two highest depletion 3rd generation CR 99 control rods will be visually inspected on all eight surfaces on each control rod.

- For 3rd generation CR 99 control rods inserted in the opposite lattice type as the lead depletion units, the two (2) highest depletion control rods shall be visually inspected during refueling outages where the control rods exceeds 90% of design NEOL. These visual inspections shall consist of an inspection of all eight faces of the control rod. For the purposes of this surveillance program, D and S lattice applications are considered equivalent, since the geometry of the absorber holes and absorber pins are identical. For example, if the lead depletion control rods are in a D or S lattice plant, inspection of the lead C lattice 3rd generation CR 99 control rods shall be performed during the outages where the depletion exceeds 90% of the design nuclear life. Conversely, if the lead inspection 3rd generation CR 99 control rods are in a C lattice plant, additional inspections of D and S lattice 3rd generation CR 99 control rods shall be performed during outages where the depletion exceeds 90% of the design nuclear life.

- Should a material integrity issue be observed, Westinghouse will arrange for additional inspections, if necessary, to determine root cause and if appropriate, recommend a revised lifetime limit to the NRC based on the inspections and other applicable information available.

- Westinghouse will report the results of the visual inspections of the 3rd generation CR 99 control rods to the NRC within 12 months of when they were performed.

Table 8-1. Operational Critical Attributes for CR 99 Designs

Operational Critical Attribute	D–Lattice CR 99 Value	C–Lattice CR 99 Value	S–Lattice CR 99 Value
Nominal wing thickness			
Maximum button thickness			
Maximum wing span			
Maximum velocity limiter diameter (with rollers installed)			
Nominal weight			
Overall length			
Velocity limiter/coupling (socket) design			
Handle design			
Envelope			

a,c

Table 8-2. Operational Criteria

Criterion	Conformance Methods(s)[1]	CR 99 C-Lattice Evaluation Results	CR 99 D- and S-Lattice Evaluation Results
(OP-1) The Westinghouse control rod socket shall be compatible with the existing CRD coupling device (spud).	Experience Testing	Extensive database of experience has shown that the design meets this criterion, i.e., the control rod couples with the spud, does not decouple inadvertently, and can be removed without problems. (meets criterion)	Extensive database of experience has shown that the design meets this criterion, i.e. the control rod couples with the spud, does not decouple inadvertently, and can be removed without problems. (meets criterion)
(OP-2) The Westinghouse control rod weight shall be similar to nominal weight of OEM blades.	Testing Analysis	[ ]a,c (meets criterion)	[ ]a,c (meets criterion)
(OP-3) The Westinghouse control rod shall be compatible with existing fuel, fuel channels, and core internals.	Experience Testing Analysis	Extensive database of experience has shown that the design meets this criterion, i.e., does not impact normal operation and scram times, does not damage surrounding fuel channels, and fits in the core internals. (meets criterion)	Extensive database of experience has shown that the design meets this criterion, i.e. does not impact normal and scram times, does not damage surrounding fuel channels, and fits with the core internals. (meets criterion)
(OP-4) The Westinghouse control rod shall be compatible with control rod handling equipment.	Experience	Extensive database of experience has shown that the design meets this criterion, i.e., all utilities installing the CR 99 design have been able to handle the rods without difficulty. (meets criterion)	Extensive database of experience has shown that the design meets this criterion, i.e., all utilities installing the CR 99 design have been able to handle the rods without difficulty. (meets criterion)
(OP-5) The Westinghouse control rod free fall velocity shall be consistent with the design basis velocity.	Experience Testing	[ ]a,c (meets criterion)	[ ]a,c (meets criterion)

Table 8-2. Operational Criteria (cont.)

Criterion	Conformance Methods(s)[1]	CR 99 C-Lattice Evaluation Results	CR 99 D- and S-Lattice Evaluation Results
(OP-6) The Westinghouse control rod shall not adversely affect scram times and settling capability in the reactor.	Experience Testing Analysis	Extensive data base of experience has shown that the design meets this criterion, i.e., scram times for Westinghouse control rods are within the experience base (and meet Technical Specification times) of the reactors into which they have been installed. (meets criterion)	Extensive data base of experience has shown that the design meets this criterion, i.e., scram times for Westinghouse control rods are within the experience base (and meet Technical Specification times) of the reactors into which they have been installed. (meets criterion)
(OP-7) Flow-induced vibration of the Westinghouse control rods shall not cause detrimental fretting of the rod or fuel channels.	Experience Analysis	Extensive data base of experience has shown that the design meets this criterion, i.e., no fretting or wear on the control rods or fuel have been seen during examination. (meets criterion)	Extensive data base of experience has shown that the design meets this criterion, i.e., no fretting or wear on the control rods or fuel have been seen during examination. (meets criterion)
(OP-8) MEOL for all new Westinghouse control rod designs shall be greater than or equal to the NEOL.	Inspection Analysis	See subsection 8.3.8 (meets criterion)	See subsection 8.3.8 (meets criterion)

Note:

1. See Section 4.2 for a discussion on Conformance Methods.

a,c

Figure 8-1. Control Rod Tolerance Envelope D-Lattice, Base Design

a,c

Figure 8-2. Control Rod Tolerance Envelope C-Lattice, Base Design

a,c

Figure 8-3. Control Rod Tolerance Envelope S-Lattice, Base Design

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