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Revision 1

October 2009

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

Westinghouse

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Revision 1: October 2009
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EXECUTIVE SUMMARY

Optimization of design dimension together with updated calculations have shown that Westinghouse CR 99 BWR control rods can be operated to significantly higher Mechanical End of Life (MEOL) conditions, as well as to Nuclear End of Life (NEOL) conditions with MEOL exceeding NEOL for all service conditions. This report (Rev. 1) 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} The individual changes included in this revision are described and summarized in the follow section.

SUMMARY OF CHANGES

Revision 1 to WCAP-16182-P-A incorporates the following list of changes:

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]a,c

1 PURPOSE

The purposes of WCAP-16182-P-A, Revision 1 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, ensures that the design requirements are met. These revised design requirements and criteria, together with those previously approved in WCAP-16182-P-A, Rev. 0, form a set of design bases for Westinghouse control rods for use in BWRs.

2. Provide updated methodology to evaluate the CR 99 design against the measurable criteria to ensure that the design meets the design bases for Westinghouse control rods used in BWRs.

3. Design Stress Limits - This revision provides an update [

   ]^{a,c}

4. Dimensional optimization together with updated calculations have shown that CR 99 can be operated to significantly higher Mechanical End of Life (MEOL) and Nuclear End of Life (NEOL), thus this report provides the justification and bases for extended life for Westinghouse CR 99 control rods used in boiling water reactors (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 Manufacturer's (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 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 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.

With respect to important factors, the CR 99 design presented in this report is the same as the CR 82 design approved by the NRC in References 3, 5, and 7, 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 30 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 design 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

CR 99 is a control rod design whose critical attributes are presented in Sections 5 through 8 of this report. A large database 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 general design requirements for Westinghouse 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 rod fulfill applicable codes and standards, including applicable parts of the ASME Boiler and Pressure Vessel Code.

Given the above design requirements, a set of measurable criteria is established which, if met, ensures 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 burnup.	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 Boiler and Pressure Vessel Code.	ME-2, 3

Notes:

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 EPRI project identified a non-cobalt material, Inconel 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 10 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 occurred (Reference 13).

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.

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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 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 to 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 does 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).

The investigation of dose impact of a new material may only involve verification that the new material contains less dose causing material (e.g., cobalt) than does the material which it is replacing. For less obvious materials 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 does 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 Section 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 be 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

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 Hole Diameter

Hole diameter directly impacts the wall thickness to the face of the blade. In conjunction with hole pitch, it impacts ligament thickness to the adjacent hole. In conjunction with hole pitch and hole depth, this parameter impacts total rod worth.

Thus, it can be seen that selection and control of this parameter are important to control rod design and in-reactor performance with respect to both mechanical and nuclear performance.

6.2.2 Hole Pitch

This parameter can affect ligament thickness between holes and total rod worth. Thus, while not as critical as hole diameter, hole pitch is still important to control rod performance.

6.2.3 Hole Depth

Hole depth is the primary parameter Westinghouse uses to control rod worth. Varying the hole depth can change the control rod worth of two otherwise identical control rods.

Due to the amount of stainless steel between the end of the hole and the inner edge of the control rod wing, and the lack of stress in that direction, differences in hole depths reasonably expected for any control rod designs for BWRs have little impact on mechanical performance.

6.2.4 Minimum Outer Wall Thickness

This parameter is important in stress analyses since any calculations done use this conservative value in determining stresses across the wall of the control rod.

During manufacture, control rods are inspected against this value to ensure that the analyses performed are valid. In general, actual values are greater than the specified minimum. Parameters which set this value include hole diameter, control rod blade wing thickness and manufacturing tolerances in the hole location.

6.2.5 Hole Ligament Thickness

This parameter is important in stress analyses done to determine stresses between holes. Parameters which set this value include hole diameter, hole pitch and manufacturing tolerances in the hole location.

6.2.6 [

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6.2.7 [

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6.2.8 Moment of Inertia

Moment of inertia is important mainly with respect to seismic behavior and ability to insert during a seismic event.

6.2.9 Mass of the Complete Control Rod

This parameter, in conjunction with the mass of the control rod without the velocity limiter and socket, is important in determining axial stresses on the control rod during scrams.

6.2.10 Mass of the Control Rod without the Velocity Limiter and Socket

This parameter, in conjunction with the mass of the complete control rod, is important in determining axial stresses on the control rod during scrams.

6.2.11 Control Rod Design Temperature

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

6.2.12 Control Rod Design Pressure

As with design temperature, design pressure is set by the design of the plant reactor coolant system. This value is used in determining the stresses across the hole walls due to differential pressures.

6.2.13 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.

6.3 MATERIALS STRENGTH PROPERTIES

Values of the parameters listed below, which are related to the material used in the control rod, are used to determine whether calculated stress levels are within acceptable ranges. Design stress is derived according to section 6.3.1 and 6.3.2.

- Young's Modulus, E
- Yield Strength, $Rp_{0.2}$
- Ultimate Strength, R_m

The values of $Rp_{0.2}$ and R_m are the minimum values specified in the material specifications.

6.3.1 Design Stress

The allowable Design Stress Limit, S_m for stainless steel, is given by:

$$S_m = \text{Min} \{ 0.9 \times Rp_{0.2}\,(T°C), 1/3 \times R_m\,(T°C) \}$$

This S_m – value should be used in all stainless steel parts of the control rod in the first place.

6.3.2 Alternative Design Stress

Alternatively, to allow previous analyses to remain valid, allowable Design Stress Limit, S_m – per Article III-2110(b) of ASME Boiler and Pressure Vessel Code Section III for stainless steel may be used:

$$S_m = \text{Min} \{ 2/3 \times Rp_{0.2}\,(20°C), 0.9 \times Rp_{0.2}\,(T°C), 1/3 \times R_m\,(20°C), 1/3 \times R_m\,(T°C) \}$$

This S_m – value may be used in all stainless steel CR – parts.

6.4 MECHANICAL CRITERIA AND DISCUSSION

Mechanical criteria to be met are stress and fatigue limits under differential static pressure, pressure cycling and scram load. [

]^{a,c} Meeting criteria specified in this section assures that applicable codes and standards are met. Stresses as defined below are used in the evaluation.

General Primary Membrane Effective Stress – P_m

This effective stress is derived from the average value across the thickness of a section of the general primary stresses produced by design pressure and other specified design mechanical loads, but excluding all secondary and peak stresses. The allowable value of this effective stress is applicable S_m – value at the design temperature.

Local Membrane Effective Stress – P_L

This effective stress is derived from the average value across the thickness of a section of the local primary stresses produced by design pressure and other specified design mechanical loads, but excluding all secondary and peak stresses. The allowable value of this effective stress is 1.5 times applicable S_m – value.

Primary Membrane (General or Local) Plus Primary Bending Effective Stress – $P_m \pm P_b$ or $P_L \pm P_b$

This effective stress is derived from the highest value across the thickness of a section of the general or local primary stresses plus primary bending stresses produced by design pressure and other specified design mechanical loads, but excluding all secondary and peak stresses. For solid rectangular sections, the allowable value of this effective stress is 1.5 times applicable S_m – value.

The following criteria are shown in Table 6-2 along with the conformance method(s) that show the criteria are met.

6.4.1 Mechanical Criterion 1 (ME-1)

Criterion

Stresses on the Westinghouse control rod handle due to normal loading and handling shall not exceed allowable values for Level A service condition (Reference 16) anytime in life.

Discussion

This Criterion ensures that the 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 the stress analysis, this load is conservatively chosen as a concentrated force at the weld location on the horizontal part of the handle. Figure 6-1 shows an example of the Finite Element Model of a double handled C-Lattice Westinghouse control rod. The applied force is assumed to be:

$$0.25 \times 2 \times \text{Control Rod Weight (in air)}$$

where:

0.25 = one fourth part of the handle (This value amounts to 0.5 for single handle designs)
2.0 = dynamic lifting factor (including a safety factor)

The maximum bending effective stress is then calculated on the horizontal part of the handle close to location of the applied load.

The maximum resulting effective stresses (P_m+P_b) must be lower than the corresponding allowable stresses. For the handle's material at 85°C, the allowable stress is $n \times 1.5\ S_m$, where n is the applicable welding factor according to Reference 16, Table NG 3352-1.

6.4.2 Mechanical Criterion 2 (ME-2)

Criterion

Stresses in the Westinghouse control rod wings due to pressure differences (ΔP) across the walls shall not exceed applicable design values as per this report any time in life. Fatigue in the Westinghouse control rod wings due to pressure differential cycles (ΔP_{cycle}) across the walls shall not exceed allowable ASME values anytime in life.

Discussion

This criterion ensures that allowable stress limits are met with the maximum outside to inside ΔP at beginning of life and maximum inside to outside ΔP at the end of life; and all differential pressures in between throughout the complete lifetime of any Westinghouse control rod design.

6.4.2.1 Pressure Difference Determination

During reactor operation, the gas pressure in the control rod blades will increase with ^{10}B depletion from the initial filling gas pressure to the design pressure at MEOL, and thus gradually change the differential pressure, ΔP, to its maximum across the walls of the blades. The differential pressure for which the blade stresses must be calculated is also a function of reactor temperature and system pressure.

Gas Pressure Buildup

[

]a,c

[

]a,c

Pressure Due to He Gas Remaining from Fabrication

[

]a,c

[

]a,c

Total Gas Pressure Build-up

The total gas pressure in the blade is calculated according to:

$$P_{TOT} = P_{He} + P_{fill} \qquad (6.5)$$

[ ]a,c

Design Internal Rod Pressure

[

]a,c

Total Differential Pressure

[

]^{a,c}

6.4.2.2 Stress Determination

[

]^{a,c}

The highest stresses caused by this ΔP occur (1) in the ligaments between absorber holes, (2) in the outer wall of a blade adjacent to a section through an absorber hole and (3) in the control rod's outer edge, farthest from the centerline of the control rod. No stresses exceed allowable limits at MEOL.

Due to the complicated geometry of the control rod, a three-dimensional FEM consisting of 20 node solid tetrahedral or brick elements is used. An example of this model is shown in Figure 6-4. The model utilizes symmetry features of the blade section. In the model, all parameters are conservatively chosen, e.g. most unfavorable tolerances. The calculations are carried out with the aid of a general purpose finite element computer program such as ANSYS (Reference 21).

[

]^{a,c}

[

]a,c

Stresses due to Pressure Loads in the Control Rod Blade

[

]a,c

Blade Outer Wall Calculation

[

]a,c

Edge Outer Wall Calculation

[

]a,c

Ligament Calculations

[

]a,c

All the calculated stresses at 300°C reactor design temperature must be lower than the corresponding allowable stress limits discussed in Section 6.4.

6.4.2.3 Fatigue Calculation

During operation of the reactor, the gas pressure in the control rod blades will increase mainly due to helium release from the boron carbide, and thus gradually will change the pressure difference across the absorber hole walls. Furthermore, normal start-up and shutdown of the reactor results in more rapid variations of the differential pressure over the walls in the control rod blades.

Load cycling

[

]a,c

[

]a,c

6.4.3 Mechanical Criterion 3 (ME-3)

Criterion

Stresses and fatigue in Westinghouse control rods due to scram induced loads shall not exceed allowable values.

Discussion

This criterion ensures that applicable stress limits according to section 6.3 of this report are met with any plant specific scram load throughout the lifetime of any Westinghouse control rod design.

6.4.3.1 Scram Load

Scram loads are given in Reference 22. During a reactor scram, the rods are hydraulically inserted in the reactor core and hydro-dynamically slowed at the end of the stroke. A scram load cycle is thus defined as a compressive scram force (acceleration) followed by a tensile scram force (deceleration). The maximum axial force in the velocity limiter and the socket occurs during the deceleration phase of the scram with cold reactor conditions, and assuming a failed buffer. This scram is considered a Level B service load and Level B service Limits as per Table 6-2 shall not be exceeded.

Scram of the reactor during the cold condition (85°C) is called cold scram, while reactor scram during normal reactor operation (300°C) is called hot scram. A "normal" scram at hot or cold conditions is considered a Level A event.

6.4.3.2 Forces and Stresses in the Velocity Limiter and the Socket

[

]a,c

[

]^{a,c}

6.4.3.3 Fatigue Calculation in the Velocity Limiter and in the Socket

[

]^{a,c}

Membrane stresses (P_m) ensuing from tensile and compressive scram forces are calculated. The alternating stresses are calculated as:

$$S_{alt} = K_t \cdot P_m \frac{E}{E_T} \tag{6.10}$$

[

]^{a,c}

Finally, the cumulative usage factor, U is calculated by:

$$U = \frac{n_1}{N_1} + \frac{n_2}{N_2} \tag{6.11}$$

where:

n1, n2	=	number of cold and hot scrams ([ ]^{a,c}, respectively)
N_1, N_2	=	number of the cold and the hot scrams to failure, respectively

The total cumulative usage factor must be less than 1.0. [ ]^{a,c}

6.4.3.4 Combined Stress Determination in the Absorber Blade

It is assumed that a scram may occur at any time during reactor operation, that is, at both cold and hot conditions. Scram stresses occur in the blade wall in a section adjacent to an absorber hole, and thus must be superimposed on the pressure induced stresses for the operation condition analyzed.

Detailed Combined Stresses Analysis of the Control Rod Blade

[ ]^{a,c}

[

]^{a,c}

Blade Outer Wall Calculation

[

]^{a,c}

Edge Outer Wall Calculation

The highest local primary membrane effective stress (P_L) and local primary membrane plus bending effective stress (P_L+P_b) across the thickness of the outer wall of the edge are calculated by the detailed FE analysis.

Ligament Calculations

The maximum effective stress in a ligament is determined by the detailed FE analysis. This stress is the highest value across the thickness of a ligament of the local primary effective stress (P_L) and local primary stress plus primary bending effective stress (P_L+P_b).

[

]^{a,c}

6.4.3.5 Fatigue Calculation for the Absorber Blade

Fatigue calculations are performed for the absorber blade under scram loads for both cold and hot scrams.

In the fatigue calculations, the following assumptions are made when calculating fatigue damage:

[

]^{a,c}

The total cumulative usage factor must be less than 1.0. [

]^{a,c}

6.4.4 Mechanical Criterion 4 (ME-4)

Criterion

Calculated stresses in Westinghouse control rod wings due to [ ]a,c shall not exceed values known to cause cracking.

Discussion

This criterion helps ensure that Westinghouse control rods reach end-of-life before the onset of cracking.

[

]a,c

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

[

]a,c

6.4.4.1 [

]a,c

[

]a,c

6.4.4.2 Differential Thermal Expansion

Thermal expansion is calculated from the information on the temperature field in the bodies involved. Thermal expansion proceeds according to the equation below.

[

]a,c

6.4.4.3 [

]a,c

[

]a,c

Example

[

]a,c

[

]a,c

[

]^{a,c}

6.4.5 Mechanical Criterion 5 (ME-5)

Criterion

The Westinghouse control rod shall be capable of insertion into the core without structural damage in the presence of an oscillatory fuel (channel) deflection of [ ]^{a,c}.

Discussion

This criterion ensures that Westinghouse control rods are capable of insertion into the core in the unlikely event of relatively large earthquake induced oscillations of fuel channels (bundles). The rod must not be too stiff to adapt to the oscillating core during insertion.

Seismic behavior in terms of insertion time in an oscillating core is essentially determined by the specific bending stiffness and moment of inertia (MOI) of the control rod. The bending stiffness is a function of the blade span, the blade thickness, hole diameter and pitch. Other factors that affect the bending stiffness are the presence of hafnium pins.

Acceptable seismic behavior of the Westinghouse CR 85 control rod design [ ]^{a,c} and its capability to withstand seismic forces have been verified in Toshiba laboratory tests under simulated earthquake conditions (Reference 23). The seismic condition was simulated by oscillating the center of the four surrounding fuel channels. In addition, a misalignment between components was also introduced. Scram insertion time was measured for different channel deflection amplitudes, up to [ ]^{a,b,c}. The tests were performed at full operating pressure and temperature. Test results are shown in Figures 6-5 and 6-6 for BWR 2/3/4/5 which present time to 90% insertion as a function of channel deflection amplitude. Figure 6-7 shows test results for BWR-6 which presents time to 75% insertion as a function of channel deflection amplitude. As Figures 6-5 to 6-7 indicate, the Westinghouse control rod blade inserts for mid-span deflections according to Table 6-2.

Inspection of the control rod after the seismic test showed that there was no functional damage and no large deformation. This demonstrates that the control rod can withstand even extremely strong seismic forces.

The Westinghouse base design of control rod blades with drilled holes in solid plates implies a consistent rod stiffness in the beam mode. That is, the expected seismic behavior is the same for rods for the C-, D- and S-Lattices. [

]a,c

[

]a,c

[

]a,c

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

a,c

Mechanical Critical Attribute	D–Lattice CR 99 Value or Range	C–Lattice CR 99 Value or Range	S–Lattice CR 99 Value or Range

Table 6-1 (cont.) **Mechanical Related Critical Attributes for CR 99 Designs. Example: AISI 316 Stainless Steel Sheet material, t < 0.6 in. (15 mm).**

a,c

Mechanical Critical Attribute	D–Lattice CR 99 Value or Range	C–Lattice CR 99 Value or Range	S–Lattice CR 99 Value or Range

Table 6-2 Mechanical Related Critical Attributes for CR 99 Designs

Criterion	Conformance Method(s)[1]	D-Lattice Reference 24	C-Lattice Reference 24	S-Lattice Reference 24
(ME-1) Section 6.4.1 Handle: Max Stress Intensity (P_m+P_b) n = 0.65 for double handle n = 1.0 for single handle				
(ME-2) Section 6.4.2 Control Rod Blade Wings : Primary Membrane Effective Stress (P_m), Local Membrane Effective Stress (P_L) and Local Membrane plus Primary Bending Effective Stress (P_L+P_b) Cycles to Failure, CF > 200				
(ME-3) Section 6.4.3 Velocity Limiter and Socket: Primary Membrane Stress Intensity (P_m) at cold (85°C) and hot (300°C) conditions Fatigue usage factor U < 1				
(ME-3) Section 6.4.3 Control Rod Blade Wings: Primary Membrane Effective Stress (P_m) at cold (85°C) and hot (300°C) conditions Local Membrane Effective Stress (P_L), and Local Membrane plus Primary Bending Effective Stress (P_L+P_b) at 85°C and 300°C Fatigue usage factor U ≤ 1				

a,c

Table 6-2 (cont) Mechanical Related Critical Attributes for CR 99 Designs

Criterion	Conformance Method(s)[1]	D-Lattice Reference 24	C-Lattice Reference 24	S-Lattice Reference 24
(ME-4) Section 6.4.4 B_4C pin to hole wall gap: Initial gap wide enough to prevent hard contact due to swelling before MEOL				
(ME-5) Section 6.4.5 Control rod insertion into the core during a seismic event without structural damage with an oscillary fuel (channel) deflection of [ ]a,c				

a,c

Note:

1. See Section 4.2 for a discussion on Conformance Methods.
2. CR 99 Stress Analysis References 24 and 25 are updated by Reference 34.

a,c

[ ]

Figure 6-1 FEM Model of Handle

a,c

Figure 6-2 Helium Release vs ^{10}B Depletion

a,c

Figure 6-3 Design Pressure Curve

a,c

Figure 6-4 FE Model of a Section of the Blade Wing Structure

(Figure Not Used)

Figure 6-5 Not used.

a,b,c

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

a,b,c

Figure 6-7 Seismic Scram Insertion Test, C-Lattice

a,b,c

Figure 6-8 Seismic Scram Insertion Test, S-Lattice

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

where:

Δ SDM is the change in SDM, relative to an original equipment manufacturer's (OEM) rod
Δk_{COLD} is the PHOENIX single bundle cold clean rod worth of the OEM rod.

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

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

$$RWD = \frac{\Delta k(\text{West.}) - \Delta k(\text{OEM})}{\Delta k(\text{OEM})} \cdot 100 (\%)$$

[

]^{a,c}

For multiple absorber control rods, total SDM is a weighted sum of the various zones. [ ]^{a,c} For Revision 1 to this report, the example CR 99 absorber material outline provided in Reference 14 has been updated and is supplemented by new Reference 33. The total SDM change would continue to be determined by (Reference 28, which has similarly been updated by new Reference 35):

[ ]^{a,c}

where:

Δ SDM_{Total} is the total change in shutdown margin and

Δ SDM_{Top}, Δ SDM_{Mid} and Δ SDM_{Bot} are the shutdown margin changes in the top, mid and bottom zones respectively.

W_{Top}, WMid, and W_{Bot} are weighting factors that describe the axial flux distribution, as discussed in Section 7.2.1 above.

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 (NEOL)

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 worths 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 design parameter for the mechanical design of the CR 99 is [

]a,c

References 28-30, updated and supplemented by new References 35-37, show calculated NEOL's 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:

$$SDM_{Westinghouse} \geq [\quad]^{a,c}\ SDM_{Replaced}$$

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 Section 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 Nuclear End-of-Life (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 an unacceptable crack.

[

]a,c

Table 7-1 Physical Related 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			
Nuclear End of Life (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 Nuclear End-of-Life (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 Section 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 can not 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 30) 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 also 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 (MEOL) for all new Westinghouse control rod designs should be greater than or equal to the Nuclear End-of-Life (NEOL).

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 uncracked 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 burn ups by performing inspections. From these inspections, guidelines for operation and the need for further inspections of the various designs are formulated.

A prototype CR 99 control rod has been operated in the Swedish Oskarshamn 3 BWR to almost 5 snvt [

]^{a,c} Thus, the criterion of a MEOL that exceeds the NEOL is considered to be met.

Table 8-1 Operational Related 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 database 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 database 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 database 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 database 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) Mechanical End-of-Life (MEOL) for all new Westinghouse control rod designs shall be greater than or equal to the Nuclear End-of-Life (NEOL).	Inspection Analysis	See Section 8.3.8 (meets criterion)	See Section 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

9 REFERENCES

1. Topical Report, Performance Verification of an Improved BWR Control Rod Design, TR BR 82-98, Revision 1, May 5, 1983 (proprietary).

2. Topical Report, ASEA-ATOM BWR Control Rods for US BWRs, TR UR 85-225, October 1985 (proprietary).

3. Letter, H. N. Berkow (NRC) to E. Tenerz (ASEA-ATOM), Subject: Acceptance for Referencing of Licensing Topical Report TR UR 85-225, "ASEA-ATOM Control Rods for US BWRs," February 20, 1986.

4. Supplement 1 to TR UR 85-225A, ASEA-ATOM Control Rods for US BWRs, October 1987 (proprietary).

5. Letter, A. C. Thadani (NRC) to E. Tenerz (ASEA-ATOM), Subject: Acceptance as a Reference Document of Supplement 1 to Topical Report TR UR 85-225, "ASEA-ATOM Control Rods for US BWRs," May 5, 1988.

6. Supplement 2 to TR UR 85-225A, ASEA-ATOM Control Rods for US BWRs.

7. Letter, A. C. Thadani (NRC) to E. Tenerz (ABB ATOM), Subject: Acceptance of Supplement 2 to Topical Report UR-85-225A, "ASEA-ATOM Control Rods for US BWRs as a Reference Document," August 8, 1989.

8. ABB Report BKE 95-044, B Rebensdorff, "Data to support the replacement of Type 304L SS with Type 316L SS as blade material in ABB BWR control rods," March 13, 1995.

9. ABB Report BX 90-37, Meeting with the NRC Regarding Cracks in the Dresden 3 Control Rods, March 27, 1990 (proprietary).

10. Letter, ABB-90-520, J. Lindner (ABB) to R. C. Jones (NRC), Subject: Meeting with NRC Regarding Update on Results of ABB Atom BWR Control Rod Inspections and Summary of Forthcoming Actions, December 14, 1990.

11. Letter, ATOF-91-130, J. Lindner (ABB) to L. Phillips (NRC), Subject: Meeting Between NRC and ABB Atom Regarding Control Rod Inspection Results at Millstone, May 29, 1991.

12. Letter, ATOF-91-273, J. Lindner (ABB) to L. Phillips (NRC), Subject: Meeting Between NRC and ABB Atom Regarding Inspection Results and Service Life Guidelines for CR-82 Rods in US Plants, December 27, 1991 (proprietary).

13. Westinghouse Atom Report, BTK 02-104, A. Lundén, Control Rods without Buttons, Nov. 2002.

14. Westinghouse Atom Report, BF 03-0225, Mechanical Design Report CR 99 Control Rods for S-Lattice BWR6, March 2003 (proprietary).

15. ABB Report RM 88-1014, The ABB ATOM Computer Model BKM-CRUD (proprietary).

16. ASME Boiler and Pressure Vessel Code, Section III, Division 1, Edition 2002 (ASME III).

17. A Strasser et al, Control Rod Materials and Burnable Poisons, EPRI Report NP-1974, Palo Alto, 1981.

18. ABB Atom Report BUA 97-023 Rev. 1, S. Eriksson, Results from Post Irradiation Examination of Absorber Material Containing Boron Carbide, 1997 (proprietary).

19. Westinghouse Atom Report BTA 03-118, G Eriksson, Calculation Methodology in predicting Pressure Buildup and Swelling of HIP Boron Carbide Absorber in Westinghouse BWR Control Rod CR 99 for US Reactors with C-, D- and S-Lattice, 2003 (proprietary).

20. Westinghouse Report BTA 03-047, G Eriksson, Stress Analysis of Westinghouse BWR Control Rod CR 99 for BWR-6 Reactors with S-Lattice, 2003 (proprietary).

21. ANSYS Inc. USA.

22. Westinghouse Atom, Report BTA 03-082, G Eriksson, Westinghouse BWR Control Rods for US Reactors. Scram Forces to Be Used in Stress and Fatigue Analysis, May 2003 (proprietary).

23. Westinghouse Report BTA 01-080, L. Plobeck, Test of Westinghouse Atom Control Rods for BWR, Normal and Seismic Scram Insertion, 2001 (proprietary).

24. Westinghouse Report BTA 03-136, G Eriksson, Stress Analysis of Westinghouse BWR Control Rod CR 99 for BWR-2/3/4 Reactors with D-Lattice, Delta Report, 2003 (proprietary).

25. Westinghouse Report BTA 03-137, G Eriksson, Stress Analysis of Westinghouse BWR Control Rod CR 99 for BWR-4/5 Reactors with C-Lattice, 2003 (proprietary).

26. Westinghouse Atom Report BCM 98-031, Validation of PHOENIX-4 against Critical Experiments, March 1998 (proprietary).

27. ABB Report UR 87-052, "XYBDRY – Control Rod Nuclear Lifetime Calculations," February 26, 1987 (proprietary).

28. Westinghouse Atom Report BTF 03-059, Nuclear Design Characteristics of Westinghouse Atom Control Rod CR 99 for BWR6 S-Lattice Reactors, April 2003 (proprietary).

29. Westinghouse Atom Report BTF 03-139, Nuclear Design Characteristics of Westinghouse Atom Control Rod CR 99 for BWR2/3/4 D-Lattice Reactors, August 2003 (proprietary).

30. Westinghouse Atom Report BTF 03-202, Nuclear Design Characteristics of Westinghouse Atom Control Rod CR 99 for BWR4/5 C-Lattice Reactors, October 2003 (proprietary).

31. ABB Report UR 87-102 Rev. 1, "ASEA-ATOM Control Rods for BWR 2/3/4/5/6 Service Limit Recommendations," April 15, 1987 (proprietary).

32. Westinghouse Atom Report, BTA 02-154, L Plobeck, Test of Westinghouse atom Control Rods for BWR Dropping Speed, Oct. 2002 (proprietary).

33. Westinghouse Report, BTK 06-1597, Mechanical Design Report CR 99 Control Rods for S-Lattice BWR6, 2007 (proprietary).

34. Westinghouse Report BTM 09-0624, G. Eriksson, BWR Control Rod CR 99 for BWR/2-4 and BWR/6 Reactors with D- and S-lattice. Mechanical End of Life prediction and Stress analysis, 2009. (proprietary).

35. Westinghouse Report BTF 06-1583, Nuclear Design Characteristics of Westinghouse Control Rod CR 99 for BWR6 S-Lattice Reactors, 2007 (proprietary).

36. Westinghouse Report BTF 06-1584, Nuclear Design Characteristics of Westinghouse Atom Control Rod CR 99 for BWR2/3/4 D-Lattice Reactors, 2007 (proprietary).

37. Westinghouse Report BTF 06-1623, Nuclear Design Characteristics of Westinghouse Atom Control Rod CR 99 for BWR4/5 C-Lattice Reactors, 2007 (proprietary).

38. WCAP-16182-P-A, Revision 0, "Westinghouse BWR Control Rod CR 99 Licensing Report," March 2005 (proprietary).