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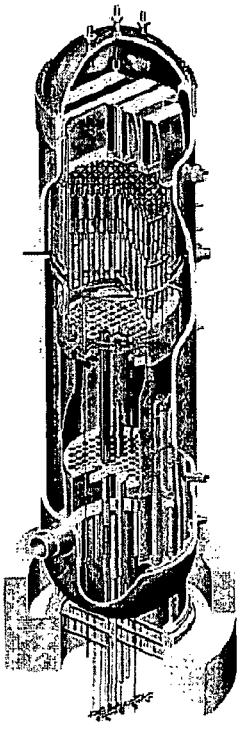
BWRVIP-217NP: BWR Vessel and Internals Project

Access Hole Cover Repair Design Criteria



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BWRVIP-217NP: BWR Vessel and Internals Project

Access Hole Cover Repair Design Criteria

1019067NP

Final Report, July 2009

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This report describes research sponsored by the Electric Power Research Institute (EPRI) and its BWRVIP participating members.

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BWRVIP-217NP: BWR Vessel and Internals Project, Access Hole Cover Repair Design Criteria. EPRI, Palo Alto, CA: 2010. 1019067NP.

REPORT SUMMARY

The Boiling Water Reactor Vessel and Internals Project (BWRVIP), started in June 1994, is an association of utilities focused exclusively on BWR vessel and internals issues. This BWRVIP report documents criteria that can be used to design a repair for the access hole covers in BWRs.

Background

In the event that significant degradation is observed in BWR access hole covers, repair to those components may be required. Utilities need criteria that can be used in the development of those repair designs.

Objectives

- To compile the appropriate design criteria for access hole cover repairs into a document that can be used by utility personnel performing the design and that could be submitted to appropriate regulatory agencies for approval of the generic design process

Approach

The contractor assembled a draft document that discussed all elements that should be considered in designing an access hole cover repair. Items discussed included:

- Design objectives
- Structural evaluation
- System evaluation
- Materials, fabrication, and installation considerations
- Required inspection and testing

BWRVIP utility representatives as well as third-party contractors reviewed the resulting draft in depth. The final report incorporates comments received during those reviews.

Results

The document provides general design acceptance criteria for the repair of access hole covers. Repairs designed to meet these criteria will maintain the structural integrity of the component under normal operation as well as under postulated transient and design basis accident conditions.

EPRI Perspective

The criteria listed in this report define a standard set of considerations that are important in designing an access hole cover repair. It is intended that these criteria will be submitted to the U.S. Nuclear Regulatory Commission and, possibly, non-U.S. regulators for their approval.

Regulatory acceptance of these generic design criteria will significantly reduce the utility effort that is required to obtain approval for plant-specific repairs.

Keywords
Boiling water reactor
Repair
Stress corrosion cracking
Access hole cover
Vessel and internals

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1
INTRODUCTION

1.1 Background

The BWR Vessel and Internals Project (BWRVIP) has prepared a safety assessment of BWR internals [1]. The evaluation of the access hole cover (AHC) included consideration of the consequence of failure of an extensively cracked AHC. It was determined that inspection and evaluation procedures have a role in assuring the long-term integrity of the AHC safety functions. Subsequently, an inspection and flaw evaluation guideline [4] for the AHC was prepared by the BWRVIP to provide a prioritized inspection strategy and guidance for evaluation of flaws.

In conjunction with inspection and flaw evaluation guidelines, a repair design criteria is outlined here for cases in which repair of AHCs is warranted.

1.2 Purpose

The purpose of this document is to provide general design guidance and acceptance criteria for permanent and temporary repair of AHCs. It is expected that individual licensees and vendors will adhere to these criteria in the application of plant-specific repairs.

The issuance of this document is not intended to imply that repair of AHCs is the only viable approach to resolution of the cracking/degradation issue.

1.3 Scope

This document is applicable to General Electric BWR/3-6 plants (BWR/2 plants do not contain AHCs). Table 1-1 shows the plant configurations that were specifically evaluated in preparing this Guideline. Configuration and material information included in the guideline is based on the best information available. Plants are advised to confirm the accuracy of this information when designing repairs. Plants not listed in Table 1-1 should obtain their configuration and material information elsewhere, but are not excluded from the scope of this Guideline.

Table 1-1
Plant Configurations Evaluated

Plant Type	Plant Name
BWR/3	Millstone, Pilgrim, Monticello, Quad Cities 1, 2, Dresden 2, 3
BWR/4	Vermont Yankee,[1] Fermi 2, Hope Creek 1, Limerick 1, 2, Susquehanna 1, 2, Browns Ferry 1, 2, 3, Peach Bottom 2, 3, Brunswick 1, 2, Hatch 1,2, Cooper, Fitzpatrick, Duane Arnold
BWR/5	LaSalle 1,2, Nine Mile Point 2, Columbia, Laguna Verde 1 & 2
BWR/6	Perry 1, Grand Gulf 1, River Bend, Clinton 1, Cofrentes

1.4 Implementation Requirements

In accordance with the requirements of Nuclear Energy Institute (NEI) 03-08, Guideline for the Management of Material Issues, the recommendations in this report are considered to be "Needed" when performing a repair to an access hole cover.

[1] Vermont Yankee is a hybrid BWR/3 and BWR/4 design.

2
DEFINITIONS

2.1 Replacement

Replacement as used in the context of this document constitutes removal of components of the AHCs that are subject to cracking and installation of new components in their place. The material and design shall be resistant to the cracking mechanisms that have been experienced in these components.

2.2 Repair

Repair as used in the context of this document is a broad term that applies to actions taken to design, analyze, fabricate and install hardware that restores the structural and functional integrity of the AHC. Repairs differ from replacement primarily in that the flaws are left in place. Weld overlay, without removal of the defect, as well as removal of flaws by a qualified machining process are also considered repairs in the context of this document.

The repairs may be temporary, i.e. designed for a specified amount of time, e.g. months of operation, or permanent, i.e. designed for the remaining life of the plant.

2.3 Access Hole and Cover

The BWR3/4/5/6 shroud support includes either one or two access holes. These provided access to the lower plenum area during assembly of the reactor internal components into the reactor vessel. Figure 2-1 shows a typical access hole. At the appropriate time in the installation process these openings were permanently closed with an access hole cover (AHC). The original equipment AHC is a flat plate that closes the access opening in the shroud support plate.

2.4 Safety Analysis Report

Safety Analysis Report (SAR) is used throughout this design criteria document to refer to the current licensing document for the plant (e.g., FSAR, UFSAR, etc.).

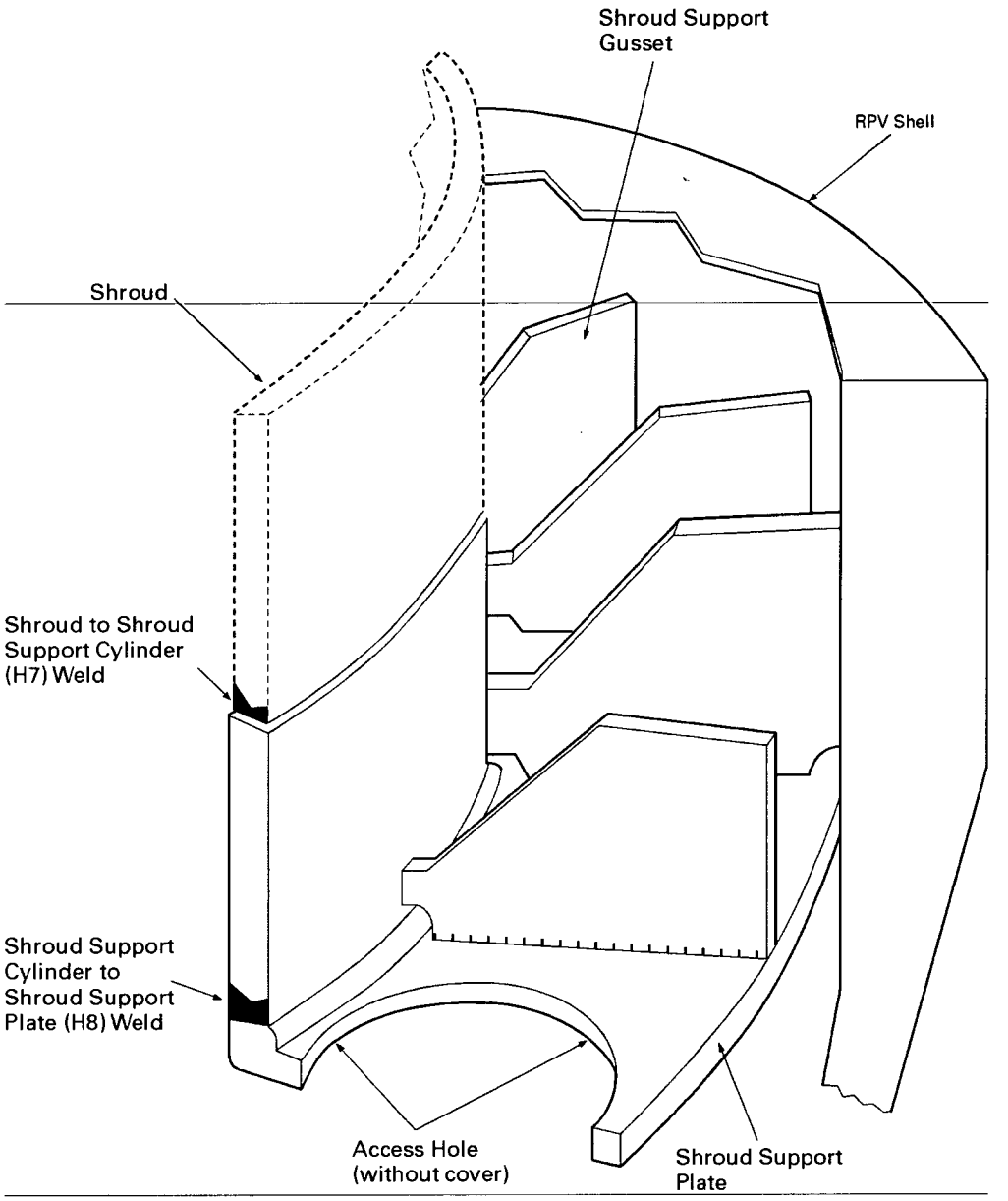

Figure 2-1
Typical Access Hole in a Gusset Type Shroud Support

3
AHC CONFIGURATION AND SAFETY FUNCTION

3.1 Generic Physical Description

The AHCs were included in BWR shroud supports to allow access to the bottom head region below the core plate during reactor internals assembly. Access was required to perform fit-up and welding of bottom head penetration components such as control rod and instrumentation housings. The access holes were subsequently welded shut with a cover after all work below the core support was complete. There are no AHCs in BWR/2 plants since the shroud support is conical and there was access for personnel to exit through the recirculation outlet nozzle in the lower plenum. This section provides a description of the different AHC designs and the features that affect cracking susceptibility.

There are several different AHC designs in BWRs. The AHC configuration for a given plant depends on the shroud support plate (SSP) design, vessel fabrication vendor, and the timing of the reactor internals assembly. For most plants there are two AHCs. Figure 3-1 is a schematic, which shows the location of the AHCs. Except at Hatch Unit 2 (and some international plants), the SSP as well as the AHCs are made of Alloy 600. The AHC is welded to the SSP using Alloy 182 or Alloy 82 weld filler material. Some sites prohibited the use of stick welds (Alloy 182) during internals installation as a fume control measure. Therefore, in many cases, the welds were made of Alloy 82. Where AHCs were made from drop-outs (thick covers), Alloy 182 was used to make the buildup because welding could be done outside the reactor vessel. Alloy 182 is susceptible to stress corrosion cracking (SCC) whereas Alloy 82 is more resistant to SCC in the BWR environment. Thus, for plant specific assessment, it is important to examine installation records to identify whether Alloy 182 or 82 was used in the attachment. For the discussion in this report, the AHC attachment welds will be referred to generically as Alloy 182/82 welds.

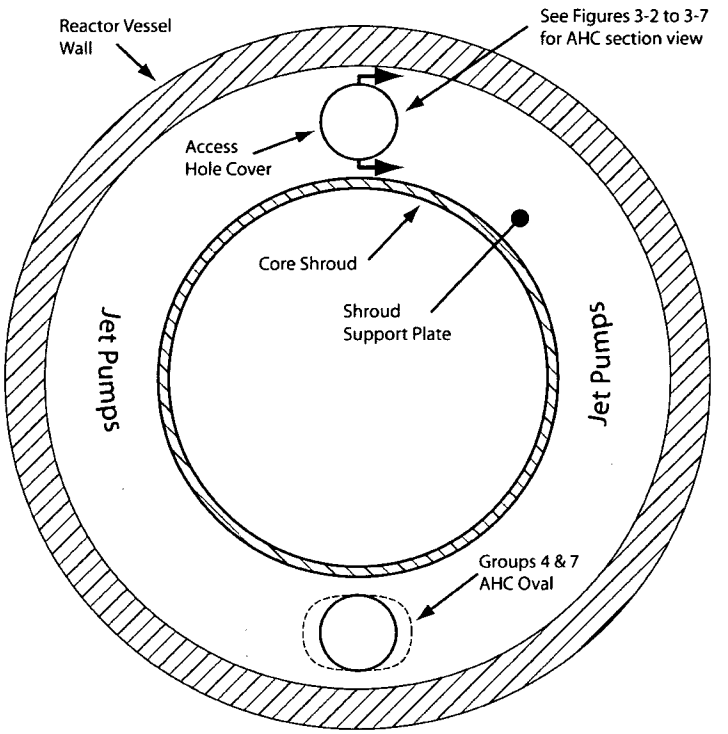

Figure 3-1
Schematic Showing Locations of Typical Access Hole Covers

In some cases, the AHC is attached to the SSP by fillet welds that are creviced. In other cases, the attachment was made using full penetration welds or using fillet welds with seal welds. The different AHC designs are classified into two broad categories – AHCs with creviced welds and non-creviced designs. Table 3-1 shows the different groups and the corresponding BWR plants that have each group. Although the plants referred to in Table 3-1 are U.S. plants (because of availability of data), the discussion in this report applies to all GE BWRs – U.S. and international – since the designs are generally similar in all the GE BWRs.

Table 3-1
Different AHC Configurations

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3.2 Creviced Alloy 600 Designs

3.2.1 Thin AHC and Conventional SSP with Ledge (Group 1)

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Figure 3-2
Thin AHC and Conventional SSP with Ledge (Group 1)

3.2.2 Intermediate Thickness AHC and Conventional SSP without Ledge (Group 2)

Figure 3-3 shows the configuration for some CE-fabricated BWR/3 vessels.

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Figure 3-3
Intermediate Thickness AHC and Conventional SSP without Ledge (Group 2)

3.2.3 Thick AHC and Conventional SSP with Ledge (Group 3)

The thick AHC uses the piece cut out from the SSP, so both the AHC and the SSP have the same thickness.

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Figure 3-4
Thick AHC and Conventional SSP with Ledge (Groups 3 and 4)

3.2.4 Thick Oval AHC and Conventional SSP with Ledge (Group 4)

The CB&I fabricated BWR/3s have the same AHC designs as described in Section 3.2.3, but with two differences.

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3.2.5 Thin AHC and Thick SSP with Ledge (Group 5)

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Figure 3-5
Thin AHC and Low Alloy Steel SSP with Ledge (Group 5)

3.3 Non-Creviced Alloy 600 AHC Designs

3.3.1 Retrofit Design for Some BWR/4 and Most BWR/5 Plants (Group 6)

Following the discovery of the first field stress corrosion cracking in creviced Alloy 600 recirculation inlet safe ends at a BWR/4 plant in 1978, efforts were initiated to create a retrofit design that eliminated the creviced Alloy 600 in the AHC design. Implementation of the retrofit design (shown in Figure 3-6) depended on the stage of vessel and internals installation in the plants under construction at that time. The retrofit designs were feasible in some of the BWR/4s and all of the U.S. BWR/5 plants.

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Figure 3-6
Retrofit Design for Some BWR/4 and BWR/5 Plants (Group 6)

Note: Retrofit design also seal welds the other AHC.

3.3.2 BWR/6 Single AHC Design (Group 7)

At the time when the retrofits were being installed in late BWR/4 and all BWR/5 plants, the BWR/6 plants (except for one overseas plant) were in sufficiently early stages of fabrication that it was possible to redesign the AHC to avoid crevices.

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Figure 3-7
BWR/6 Single Oval AHC Design (Group 7)

3.4 Safety Design Bases

The AHC forms part of the flow boundary of the shroud support plate. This flow boundary has a safety function in directing and controlling the reactor coolant flow. In the event of a recirculation line loss of coolant accident (LOCA), this core coolant boundary is required to maintain two-thirds core height coverage (refloodable volume). The AHC is not structurally important to the shroud support itself (it is not required to support the reactor core).

3.5 Event Analyses

As previously stated, the purpose of this document is to provide general design criteria for repairs of degraded AHCs. Accordingly, various events and operational conditions must be considered to ensure that the repair does not inhibit the ability of the AHC to perform the basic safety and operational functions. The following general load cases shall be considered in design of the proposed repair.

3.5.1 Normal Operation

The repair design should consider loads existing during periods of reactor startup, shutdown, and power operation. This includes dead weight of the AHC, differential pressure, and thermal-hydraulic loads (including flow induced vibration (FIV)).

3.5.2 Anticipated Operational Occurrence (Upset Conditions)

Loads due to anticipated operational occurrences, which have the potential to increase AHC loads above normal operation, should be considered. Typical events include: maximum system pressure, recirculation flow control failure (max demand), loss of feedwater with feedwater restart without feedwater heating, and inadvertent activation of a safety relief valve. Also, the combination of normal loads plus operating basis earthquake (OBE) loads is an upset event. System testing conditions shall also be considered an upset condition.

3.5.3 Design Basis Accidents (Emergency/Faulted Conditions)

Loads associated with a design basis earthquake in conjunction with a steam line or recirculation line break shall be considered as required by the design basis for the plant. All components of these loads should be considered.

Leakage into the annulus through a repaired AHC will be available to reflood the core following a steam line break. The AHC has no structural function following a steam line break LOCA. Therefore, it may be possible to justify relief from consideration of a steam line break accident. Such relief should be obtained from the regulatory authority on a plant specific basis.

3.5.4 Loading Combinations

All loads, including seismic and LOCA events, shall be combined in accordance with specific plant SAR requirements.

4
SCOPE OF REPAIRS

4.1 Scope of Repairs

These criteria apply to all of the AHC configuration groups. These criteria are not applicable to "like for like" replacement of components. Removal from the vessel to facilitate repairs of components would, however, be within the scope of this criteria document.

The AHC repairs may address cracking in IGSCC susceptible components by a number of options. Local repairs such as weld overlays or mechanical devices that leave the flaws in place but structurally replace the flawed area may be used, or a qualified machining process may remove the flaw if it is not through-wall, with subsequent weld repair if required. Repairs include the hardware necessary to connect the new components to the existing assembly.

BWR3/4/5 shroud supports were designed and constructed by the RPV fabricator (vessel vendor). In some cases the RPV Code design specification extended the Class 1 vessel boundary (jurisdiction) to include the complete shroud support assembly.

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Figure 4-1
Typical Code Boundaries Associated with AHC

5
DESIGN OBJECTIVES

5.1 Design Life

The design life of the repair components will normally be for the remaining life of the plant plus planned life extension beyond the current operating license. However, temporary repairs with a design life as specified by the plant licensee are an acceptable option.

5.2 Safety Design Bases

The repair shall be designed such that the safety bases described in Section 3.4 of this document, the plant SAR and technical specifications are met.

5.3 Safety Analysis Events

Safety analysis event scenarios described in individual plant SARs remain valid and unaltered by the criteria contained in this document.

5.4 Structural Integrity

The AHC repair shall be designed to provide structural integrity for all specified loading conditions. Thermal-hydraulic loads, including flow-induced loads, acting on the AHC for normal, upset, emergency and faulted conditions in addition to seismic loads shall be considered. The pressure differences used for these events shall be consistent with the current plant licensing basis documents.

5.5 Retained Flaw(s)

Where an existing weld or component is being structurally but not physically replaced, the repair design and analysis shall consider the entire range of postulated crack growth, from the existing flaw size to the full length of the component.

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5.6 Loose Parts Considerations

Repair hardware shall be designed to minimize the potential for loose parts inside the vessel during installation of the repair and during reactor operation. All repair hardware shall be captured and held in place for the design life of the repair by a suitable method. Threaded fasteners, which are not otherwise captured by the design, shall be lock welded or fitted with lock-welded retainers during installation. Alternatively, threaded fasteners may be locked by other suitable methods such as mechanical retainers, which have been qualified for in-reactor use. Components, which are abandoned in place, shall be evaluated for loose parts consideration. The evaluation should show by analysis and/or inspections that the components will not become loose parts or, if necessary, demonstrate the acceptability of potential loose parts.

5.7 Physical Interfaces with Other Reactor Internals

The repaired configuration shall interface properly with other reactor internal components to ensure that all components continue to function as intended. Clearance with all internal features shall be evaluated to assure that there is sufficient clearance for the planned installation of the repairs. Clearance checks shall include other implemented or planned repairs, such as shroud repairs, which add hardware in the vessel.

Consideration shall be given to minimize or avoid interference with other components and welds that may require future inspections or repairs.

When a repair covers a weld, it must either:

a. Structurally replace the weld or

b. The weld must be inspected per the I&E guidelines (remove hardware if required) or

c. Analysis must postulate any credible failure up to complete failure of the weld.

Repairs, which attach to other in-vessel components, must not adversely affect the function or structural integrity of those components.

5.8 Installation

A design objective for the AHC repair shall be to minimize the critical path in-vessel time required. This can be accomplished by using dimensionally tolerant hardware minimizing the need for as-built dimensions. Another approach would be to obtain the necessary as-built information at an outage prior to the planned installation.

5.9 AHC and Leakage

The repair shall minimize leakage through the AHC during normal and accident conditions. Some leakage from the AHC, however, is acceptable provided the design basis functions are maintained. See Section 8.1 for specific requirements related to leakage evaluations.

5.10 Design Verification

For each repair, all of the applicable requirements related to loose parts, vibration analysis, inspection, etc. shall be evaluated and satisfied to ensure that the as-installed hardware configuration is consistent with the design and analysis assumptions.

6
GENERAL DESIGN CRITERIA

Repair designs shall meet the individual plant SAR and other plant commitments for RPV internals mechanical design. Although AHCs are not part of the reactor pressure boundary, in many of the early generation BWRs they were required to use the stress allowables of ASME III as design guidance. Where commitments exist to meet the "intent" of ASME Section III or no plant specific requirements exist, Table 6-1 should be used. Table 6-1 provides a summary of the recommended ASME design guidance associated with the items that may potentially be affected by the repair.

Table 6-1
Summary of Recommended ASME Design Guidance

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7
STRUCTURAL AND DESIGN EVALUATION

The loads and load combinations used in analyzing a repair design shall be consistent with the current licensing basis for the plant. In the event that adequate information is not contained in the licensing documents, the loads and load combinations presented in Sections 7.1 and 7.3 may be used.

7.1 Load Definitions-Applied Loads

The applied loads on the reactor internals consist of the following: deadweight, differential pressure, hydraulic loads, seismic inertia, seismic anchor displacements, LOCA phenomena, SRV Opening, loads due to flow induced vibration and thermal and pressure anchor displacements. A general discussion of these loads with some specific applications to the AHC follows below.

7.1.1 Applicability of Hydrodynamic Loads

In general, hydrodynamic loads incurred due to SRV discharge, pool swell, condensation oscillation, main vent clearing and chugging are applicable to Mark II and III containment types. These loads are not significant for the vessel and internals in Mark I containment types where the torus and drywell are not dynamically coupled to a substantial degree. Also, the annulus pressurization loads may not be included in the licensing basis for Mark I containment plants.

7.1.2 Deadweight (DW)

Deadweight consists of the weight of the component, the weight of other internals supported by the component and the weight of the entrapped water, if any.

7.1.3 Hydraulic Loads (F)

The hydraulic loads arise from the fluid momentum forces. Hydraulic loads include fluid drag effects where applicable. The magnitude and source of the load is a function of operating conditions and event scenarios.

7.1.4 Differential Pressure (DP)

Differential pressures arise throughout the reactor internals as a result of forced fluid flow throughout the RPV. This includes RPV and system pressure expansion related loads including the internal pressure on piping. The pressures during the transient conditions are based on the information provided in the vessel thermal cycle diagram or other plant design-basis information.

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7.1.5 Seismic Inertia

Seismic inertia consists of horizontal and vertical inertia forces acting on the entire component due to seismic excitation of the RPV. The locations where the seismic excitation is imparted to the component are at the connection to the RPV and the shroud. (For the AHCs, this location is the shroud support plate.)

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7.1.6 Seismic Anchor Displacements

Seismic anchor displacements are applied at the attachment points of the component to the RPV or core shroud. These displacements, if available, are obtained from the RPV internals seismic analysis report. If not available, additional analysis may be required.

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7.1.7 Safety Relief Valve Opening (SRV)

SRV air clearing loads are induced by SRV actuation, which produces a rapid compression of the air mass in the interior of the SRV discharge pipes. The internal pressure drives the water out of the submerged SRV discharge device (rams head or quencher) and ejects a high-pressure air bubble into the suppression pool below the water surface, causing oscillating pressures on the suppression pool boundary. The oscillating pressures impart structural motions, which may cause dynamic excitations of the structure and contained equipment. The SRV loads are determined for the following three event scenarios:

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7.1.8 Main Vent Clearing (MVC)

Main vent clearing loads are induced during IBA and DBA by purging drywell air through the main vent and into the suppression pool. The suppression pool boundary pressures may cause dynamic excitations of the structure and the contained equipment.

7.1.9 Annulus Pressurization (AP)

Annulus pressurization refers to the loading on the biological shield and the reactor vessel following a postulated pipe rupture. The pipe break is assumed to occur at the vessel nozzle safe end to pipe weld. This rupture allows a rapid mass and energy release into the small annular region between the biological shield wall and RPV. The mass and energy released during this postulated pipe rupture causes two types of loads:

1. A short-term transient, asymmetric, differential pressure within the annular region between the biological shield wall and the RPV.
2. A reaction to the jet stream release of the RPV inventory and the impact of the ruptured pipe against the pipe whip restraint, which is attached to the biological shield wall.

7.1.10 Pool Swell, Condensation Oscillation and Chugging (PS, CO, CHG)

Pool swell (PS) loads are induced during DBA by the injection of drywell air into the suppression pool during the LOCA and the subsequent expansion of the air bubble which results in the rise of the suppression pool surface. Structures above the pool surface may experience loads. In addition to the initial impact loads, these structures may experience drag loads as water flows past them.

Condensation oscillation (CO) loads are included during an intermediate-break accident (IBA) and design basis LOCA (DBA) following vent air cleaning and pool swell (PS). There is a period of high steam flow rate through the vent system where the hydrodynamic pressure oscillations may cause dynamic excitations of the structure and contained equipment.

Main vent chugging (CHG) loads are induced during an SBA (small break accident), IBA, and DBA when there is insufficient steam flow to maintain a steady steam jet at the vent exit. A random formation of steam bubbles, which alternatively form and collapse at the vent exit, produces hydrodynamic pressure oscillations on the pool boundary for Mark II pressure suppression containments and on the weir wall and pool boundary for Mark III containments. These pressure oscillations may cause dynamic excitations of the structure and the contained equipment.

7.1.11 Flow Induced Vibration (FIV)

The flow induced vibration (FIV) loads are caused primarily by turbulent flow exciting the natural frequencies of the components.

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7.1.12 Thermal and Pressure Anchor Displacement

The anchor points of the reactor internals grow vertically and horizontally at different rates due to differences in the materials (low alloy steel for the vessel versus stainless steel for many of the internals). Also, these displacements are expected to vary during certain transients due to the differences in temperatures and pressures. The RPV temperatures during the transient conditions are based on the information provided in the vessel thermal cycle diagram. The following load cases need to be considered:

- Displacements during normal operation: NOD
- Displacements during operating transients, typically enveloped by the loss of feedwater pumps transient: LFWPD (This should be verified on an plant specific basis.)
- Displacements during a loss of coolant accident: LOCAD

Loads/stresses associated with steady state or transient conditions are also included along with the differential thermal displacement effects.

7.2 Service Level Conditions

The applicable service level conditions shall be in accordance with the individual plant SAR or other plant commitments for RPV internals mechanical design. Where commitments exist to utilize the requirements of NUREG-0800 or no plant specific guidance exists, the following descriptions may apply. These descriptions of general load combinations by service level are based on the current regulatory guidance (provided in NCA-2142.4 of ASME Section III and Appendix A of Section 3.9.3 of NUREG-0800 for U.S. plants).

7.2.1 Service. Level A (Normal Operating Conditions)

Service Level A loads should include the combination of all sustained loads that are anticipated during normal plant/system operation. These include deadweight of all supported components, differential pressures, and thermal-hydraulic loads (including FIV).

7.2.2 Service Level B (Upset Conditions)

Service Level B loads include loads due to anticipated operational occurrences that have the potential to increase the loads acting on the reactor internal components above those experienced during normal operation. Typical events include normal operation loads plus system operating transients (SOT). The SOTs shown on the RPV thermal cycle diagram should be used to determine the applicable transient conditions. Also, the combination of normal loads plus OBE loads is considered an upset event.

7.2.3 Service Level C (Emergency Conditions)

Service Level C loads include the combination of all sustained normal operation loads in conjunction with loads from the design basis pipe break (DBPB). The DBPB includes all postulated pipe breaks other than a LOCA, MSLB, or feedwater pipe break. These include postulated pipe breaks in Class 1 branch lines that result in the loss of reactor coolant at a rate less than or equal to the capability of the reactor coolant makeup system.

7.2.4 Service Level D (Faulted Conditions)

Service Level D loads include the combination of all sustained loads in conjunction with several combinations of design basis events. These combinations include the DBPB, MSLB/feedwater pipe break, or LOCA and the SSE (where applicable per the plant specific design basis). All components of these loads should be considered.

For plants that use systems for injection (e.g., jet pumps for LPCI injection and core spray injection), the loads associated with the injection are treated as a faulted condition. This assumption is acceptable provided that the system functional requirements for delivery of coolant under long-term DBA conditions are ensured.

7.3 Load Combinations

The load combinations used in the evaluation should be consistent with the requirements of the plant SAR or related licensing basis documentation. Typically, Section 3.9 of the SAR contains the necessary information on this subject including, for some plants, hydrodynamic loads (i.e., "new loads") and/or AP loads. In the event that adequate definition of load combinations is not contained in the plant licensing basis documentation, the following load combinations may be used.

Load combinations used to analyze reactor internals vary, depending on the plant vintage. There are two major categories of plants: those with Mark II or Mark III containments where hydrodynamic events in the suppression pool cause vessel internals loads, and those with Mark I containments where hydrodynamic effects in the torus do not cause significant loads on the vessel internals.

7.3.1 Mark I Plants

For the purposes of providing a general guideline in the event that load combinations are not specified in the SAR, the set of load combinations shown in Table 7-1 may be used.

Table 7-1
Load Combinations for Mark I Plants

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7.3.2 Mark II and III Plants

For Mark II and III plants, the method for load combination was specified at the time that the loads caused by hydrodynamic events were defined and labeled "new loads". A set of load combinations, in lieu of plant specific documentation, is shown in Table 7-2.

Table 7-2
Load Combinations for Mark II and III Plants

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7.4 Allowable Stresses

Allowable stresses under the above loading combinations should be consistent with the current plant SAR. Unless otherwise specified, the following allowables apply:

- Normal and Upset load combinations – Normal and Upset (Service Level A and B) code limits.
- Accident load combinations – Emergency (Service Level C) or Faulted (Service Level D) code limits apply. To insure integrity of pre-loaded mechanical joints and repair components, which are critical to meet functional requirements, it may be necessary to also evaluate the effect to secondary loadings with emergency and faulted load combinations.

The plant specific submittal shall identify the specific sections and subsections of the ASME Code utilized to designate allowable limits. See Section 6.1 for a summary of the applicable ASME criteria.

Table 7-3
Load Term Definitions for Tables 7-1 and 7-2

Loading Term	Description[1]
DW	Deadweight
F	Hydraulic loads including fluid drag were applicable
DP	Loads caused by differential pressure
FIV	Flow induced vibration loads, where applicable
NOD	Normal Operating Displacements
N	Combination of Normal Operating Loads
OBEI	OBE Inertia
OBED	OBE Anchor Displacement
OBE	Combination of All OBE Induced Loads
SSEI	SSE Inertia
SSED	SSE Anchor Displacements
SSE	Combination of All SSE Induced Loads
LFWPD	Displacements during operating transients
SOT	Loads induced by system operating transients
LOCAD	LOCA induced Anchor Displacements
SRV	Safety Relief Valve Opening
SRV_{OPN}	One or two SRV's open
SRV_{AIL}	All SRV's open
SRV_{ADS}	SRV associated with ADS
SBA	Small Break Accident Loads
IBA	Intermediate Break Accident Loads
DBA	Design Basis Accident Loads
AP	Annulus Pressurization
PS	Pool Swell
MVC	Main Vent Clearing
CHG	Main Vent Chugging
CO	Condensation Oscillation

1. Use pressures and temperatures that are applicable to the conditions associated with the operating condition and postulated event.

7.5 Consideration of Shroud Repair or Cracking

For BWR shrouds that have cracking at various circumferential welds or have had repair hardware installed, the bounding anchor displacement values and seismic response characteristics calculated, as a part of the assessment or repair hardware design analysis should be used. However, an anchor displacement other than the bounding value may be assumed on a plant-specific basis, as long as appropriate justification is provided.

7.6 Flow Induced Vibration

The repair shall be designed to address the potential for vibration, and to keep vibration to a minimum.

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7.7 Impact on Existing Internal Components

Changes in the AHC stiffness and weight, which result from repairs, shall be evaluated to demonstrate that the shroud support assembly satisfies design basis stress limits.

For repairs, which attach to other in-vessel components, the interface loading on those components must be analyzed to demonstrate that their structural integrity is maintained.

7.8 Radiation Effects on Repair Design

The design of the repair shall account for the effects of irradiation relaxation utilizing end-of-life (including life extension if applicable) fluence on the materials.

7.9 Analysis Codes

All thermal-hydraulic and structural computer codes utilized in the design analysis shall be appropriately benchmarked.

The designer may utilize new or improved calculation methods. For these techniques, appropriate benchmark information to demonstrate that the method is conservative and bounding for the application, shall be provided.

7.10 Thermal Codes

The design and analysis of AHCs shall consider the operating conditions and events specified on the original plant RPV and nozzle thermal cycle diagrams or equivalent source. The design shall assume the number of thermal cycle events equal to or greater than the number assumed in the original RPV design.

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7.11 Corrosion Allowance

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8
SYSTEM EVALUATION

8.1 Leakage

Leakage paths allow recirculation water to bypass the core and/or the steam separators. Leakage paths include cracks and repair openings in the shroud, in the jet pump diffuser adapter and in the access hole cover (AHC).

The AHC leakage evaluation shall consider the potential impact of leakage from a wide range of sources for normal and design conditions. Various leakage sources combine to provide the leakage from the lower plenum, through the AHC, into the RPV annulus. The leakage sources include retained through wall cracks in the original AHC welds or any mechanical repair joints. Previously implemented repairs or replacements of core spray internal piping, core spray spargers, core shroud, or jet pumps may have resulted in leakage paths between the annulus and the core regions. Existing but un-repaired or replaced flaws in core spray internal piping or core spray spargers might also exist at the time when a repair is implemented.

The core plate pressure drop, or pressure drop across the lower shroud, provides the driving head in calculating some of these leakage flows. The leakage evaluation shall be based on system temperature, pressure and system flow conditions that are consistent with those used in the licensing basis and/or operating conditions. The bounding structural crack configuration assumptions may not be applicable for calculation of the AHC leakage. As a guideline, crack leakage flow is very small and has minimal impact on normal plant operation. However, leakage flows may in some unique cases affect normal operation and accident evaluations.

8.1.1 Leakage Impact and Acceptance Criteria-Normal Operation

During normal operating, AHC leakage can decrease core flow. The criteria for acceptable reduction in core flow during normal operating conditions is plant specific and must be evaluated on a case-by-case basis.

8.1.2 Leakage Impact and Acceptance Criteria Accident Conditions

Post accident reflooding capability is a safety consideration where AHC leakage may exist. The AHCs are critical to maintaining two-thirds core coverage. Any fluid that leaks from the AHC assembly into the RPV annulus is potentially unavailable for core cooling. A reduction in the core cooling flow (whether as a result of leakage through cracks or for any other reason) may result in an increase in the peak cladding temperature (PCT). Thus, the tolerable AHC leakage is a function of acceptable increase in the calculated value of PCT. The PCT is a part of the plant's LOCA analysis.

8.1.3 Leakage Evaluation Methodology-Accident Conditions

The following items shall be addressed in the leakage evaluation methodology for accident conditions:

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8.2 Emergency Operating Procedure (EOP) Calculations

Inputs to the EOP calculations, such as bulk steel residual heat capacity and reduction of reactor water inventory shall be addressed based on replacement hardware mass and water displacement.

8.3 Power Uprate

For those units currently undergoing a power uprate program, the resulting increased loadings must be considered in the repair design. If a power uprate program is implemented after the installation of an AHC repair, the uprate program shall address any increased loads imposed on the repaired AHC. It shall be the responsibility of the licensee to include, as a power uprate design basis requirement, an evaluation of power effects on AHC repairs.

9
MATERIALS, FABRICATION AND INSTALLATION

9.1 Materials, Fabrication and Welding

Materials, fabrication and welding shall be in accordance with the current version of BWRVIP-84 [5]. The repair of AHCs will most likely involve machining or welding on the existing Alloy 600 shroud support or the existing AHC. In addition, modified nickel base wrought Alloy 600 type material may be an appropriate material for the fabrication of AHC repair items. BWRVIP-84 does not currently address Alloy 600 materials. Therefore, the following specialized Alloy 600 specific materials requirements are provided for AHC repair. Additionally, based on the field experience with Alloy 600, the repair needs to be evaluated to assure that it does not introduce a crevice associated with any new welds consistent with the discussion in Section 9.3. For Alloy X-750, additional requirements are also provided to address recent field experience.

Other Ni-base alloys may also be used such as Alloy 690, 718, or 725. Compositional and processing requirements (such as those given in Reference [5] for 300 series stainless, XM-19 and X-750 materials) shall be documented to substantiate the alloy's adequacy in the specific material condition used in the application.

9.1.1 Type Alloy 600 Material Requirements

The requirements for Alloy 600 fall into two categories: processing requirements associated with the original shroud support and AHC component materials and fabrication of new components that are installed as part of the AHC repair.

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9.1.2 High Strength Alloy X-750

The requirements for Alloy X-750 material that shall be used for AHC repairs is covered in Appendix B of Reference [5]. To assure that there is adequate mitigation of SCC, all applications shall use proper design practice for minimizing stress concentration at thread roots or changes in geometry. Secondly, the detailed analyses shall include all loads that could arise during normal operation such as those imparted by installation, thermal expansion and misalignment.

9.2 Part Circumference Weld Overlay Repair

Part circumference weld overlay in accordance with BWRVIP-34 [6] may be applicable to AHCs but with some limitations:

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9.3 Crevices

A crevice is a narrow region between two reactor internal surfaces, into or through which there is limited flow of reactor coolant. The crevices of greatest concern are those that involve contact between any material and the weld heat affected zone of existing 300 series stainless steel internals components, and those that involve contact between any material and any Alloy 600 material.

The design shall minimize crevices between new components, and between new components and original components to minimize the potential for crevice-induced stress corrosion cracking. It is recognized that mechanical joints, closures, and fasteners typically used in reactor internals and in repairs necessarily contain crevices. The design of such features should avoid sensitized areas, should utilize IGSCC-resistant materials, and should be vented to the extent practical to minimize stagnant conditions. BWRVIP-84 provides guidance for minimizing crevice conditions.

9.4 Pre-Installation As-Built Inspection

The repair design shall specify the as-built dimensional tolerance that the repair will accommodate. For critical measurements a pre-installation dimensional check shall be performed and reconciled with design tolerances. The repair design shall include special features, as required, to accommodate expected deviations in component characteristics such as diameter, ovality and alignment.

9.5 Post-Installation As-Built Inspection

The designer shall specify the as-built inspections required for the entire repair or replacement, commensurate with design basis considerations and Code requirements. The objective of the inspections is to verify that the repair hardware has been correctly installed. The inspections shall verify critical tolerances, hardware fit-up, alignment and anchorage installation as a minimum. Inspections can be performed in process or at the completion of the installation.

9.6 Installation Cleanliness

The design shall minimize the in-vessel debris generation. Debris recovery methods shall be incorporated into the installation process consistent with the amount and characteristics of debris

generated. All modification hardware and tooling shall be cleaned and inspected for cleanliness prior to use in the reactor. Previously demonstrated proactive measures such as FME (Foreign Material Exclusion) shall be taken to prevent loose parts or foreign material being left during the installation as discussed in Section 5.6.

If any debris will remain in the vessel after installation, the debris shall be identified and its affect on plant components and fuel shall be evaluated. This evaluation will include, as a minimum:

a. Material identification of the debris.
b. Effect of the debris on reactor coolant system (RCS) chemistry.
c. Ability of the debris to pass through the reactor core without affecting zircalloy fuel rods (deposition will affect heat transfer in the core).
d. Probability and effect of debris blocking essential systems interfacing with the RCS's (pumps, valves, etc.)
e. Effect of the debris on other essential components in the plant and an assessment of whether or not the debris will increase the probability of short-term or long-term material degradation of essential components in the plant.
f. Specific requirements of the utility's loose parts or foreign material exclusion program.

9.7 ALARA

The design should utilize construction and installation techniques that minimize the radiation exposure to the workers using ALARA practices in all steps.

It is desirable that the repair be installable remotely from the refuel floor with the vessel flooded to normal refueling levels and shutdown cooling in service if practical. The repair hardware should include features which facilitate handling during installation and during subsequent inspection, adjustment, and removal/replacement of components. The modification design shall be done in concert with the design of special installation tooling to facilitate remote installation.

The repair should minimize the amount of radwaste generated.

The repair should minimize the radiation exposure to the plant workers in future repair and inspection operations involving the newly installed structures or activities that are affected by the new structures.

9.8 Qualification of Critical Design Parameters and Process

Critical design parameters shall be identified and shall be qualified and documented to ensure that the parameters meet the design basis. Appropriate mockups shall be utilized and shall be designed to represent the configuration of the actual installation being mocked up as closely as possible. Differences between the mockup and actual installation shall be evaluated and the effect on the qualification shall be documented:

As a minimum, qualification of critical design parameters shall include:

- Preload in tensioned members
- Critical dimensions or tolerances
- EDM, Rotobroach, abrasive waterjet, or-other machining process
- Mechanical locking devices
- Welding procedures and operators.

10
INSPECTION AND TESTING

10.1 Inspection Access

The repair design shall be such that inspection of reactor internals, reactor vessel, ECCS components and repair hardware is not impaired. The AHC repair hardware shall not cover or obstruct any other system welds that are not structurally replaced by the repair hardware. The installed repair hardware shall not interfere with refueling operations and shall permit servicing of internal components.

Where practical, it may be beneficial to have an AHC repair that is removable and replaceable without requiring destructive removal. This is to provide greater access to the lower plenum area for other possible future inspections and/or maintenance/repair activities.

10.2 Pre and Post Installation Inspection

AHC welds that are structurally replaced by repair hardware shall not require pre-installation or post-installation inspection.

Existing reactor internal components utilized for repair anchorage shall be inspected prior to repair installation, as specified by the designer, to ensure the structural integrity of the anchorage for the specified design life of the repair.

If the AHC repair attaches to other AHC components, the required inspection program for those components [4] must be reviewed and, if necessary, changed to match the new function of the components. The same approach is required when the repair attaches to in-vessel components other than AHC components.

Inspections required for the entire repaired AHC assembly shall be specified commensurate with design considerations and Code requirements applicable to the specific design for the specified design life of the repair. These inspections shall be consistent with the requirements and scope of BWRVIP-180 [4].

Inspection for new welds shall be specified commensurate with the design requirements. These inspections shall be consistent with the requirements and scope of BWRVIP-180 [4].

Existing hardware that is abandoned in place shall be inspected, as specified by the designer to ensure that the structural integrity is consistent with the established design criteria for the specified design life of the repair, including that it will not become a loose part, if this is a requirement.

In this section, post installation inspections include both those inspections that are to be performed upon completion of the repair (final acceptance inspections) and those inspections that are to be performed inservice.

10.3 Testing

To ensure proper functioning of the repaired AHCs plant operators shall monitor the relationship of core thermal power to core flow.

11
QUALITY ASSURANCE PROGRAM

Repair design, fabrication and installation activities shall be conducted under a quality assurance program meeting regulatory requirements (for example 10 CFR 50, Appendix B for domestic units).

12
DESIGN BASIS DOCUMENTATION

The following documentation shall be prepared and forwarded to the plant owner and maintained as permanent records:

- *Design Specification(s)* documenting the plant specific design requirements such as pressures, temperatures, flow rates, and SAR load combinations.
- *Design Report(s) or Calculations* documenting structural and systems analyses. Where the repair affects material within the boundary of ASME Code, a design report, certified in accordance with ASME Section III Code, shall be prepared documenting reconciliation of the repair with the original ASME Code design report. A Suitability Evaluation as required by ASME Section XI for Section III Components shall also be prepared.
- An evaluation to determine whether or not regulator approval is required before implementing the proposed repair
- Installation Documentation Package per existing administrative procedures, including such items as weld procedures, qualifications, testing requirements, etc.
- QA Verification Records per existing administrative procedure, including material test reports, heat treatment records, inspection records, etc.
- Recommendations for future inservice inspection of the repaired AHC.

13
REFERENCES

1. *BWRVIP-06-A: BWR Vessel and Internals Project, Safety Assessment of BWR Reactor Internals*, March 2002. EPRI Technical Report 1006598.
2. Regulatory Guide 1.84, Design, Fabrication, and Materials Code Case Acceptability – ASME Section III Division I.
3. 10 CFR 50, Appendix B.
4. *BWRVIP-180: BWR Vessel and Internals Project, Access Hole Cover Inspection and Flaw Evaluation Guidelines*. EPRI Technical Report 1013402, November 2007.
5. *BWR Vessel and Internals Project, Guidelines for Selection and Use of Materials for Repairs to BWR Internal Components (BWRVIP-84)*. October 2000. EPRI Report 1000248.
6. *BWR Vessel and Internals Project, Technical Basis for Part Circumference Weld Overlay Repair of Vessel Internal Core Spray Piping (BWRVIP-34)*. May 1997. EPRI Report TR-108198.
7. ASME Boiler and Pressure Vessel Code, Code Case N-580-1.

A
AHC REPAIR CONCEPTS

A number of AHC repairs have been installed in BWRs. This appendix includes a discussion of the conceptual design of these repairs and lessons learned during repair installation or subsequent plant operation. Note that the NRC has not necessarily accepted the repair approaches discussed in this appendix.

Table A-1 summarizes previous U.S. BWR experience with AHC repairs. Based on review of prior AHC repair history, four distinct previous repair concepts are apparent. The important attributes of these four concepts are documented in the following section. In addition, some repair concepts intended to address cracking of the cylindrical portion of top hat (Group 6 or 7) AHCs are presented. The top hat repair concepts have not yet been implemented.

Table A-1
History of Prior (GEH) AHC Mechanical AHC Repairs

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Table A-1
History of Prior (GEH) AHC Mechanical AHC Repairs (Continued)

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A.1 4-Bolt with C-Ring Seal

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Figure A-1
Preparation of SSP for 4-Bolt (PB3) AHC

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Figure A-2
Plan View of Installed 4-Bolt AHC

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Figure A-3
Typical Detail of Installed L-Shaped Bolt

A.2 Temporary Toggle Clamp Design

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Figure A-4
Temporary (Toggle Clamp) AHC [[Content Deleted - EPRI Proprietary Information]] TS

A.3 6-Bolt with Spot Face [[Content Deleted - EPRI Proprietary Information]] TS

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Figure A-5
Preparation of SSP for 6-Bolt AHC

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Figure A-6
Plan View of Installed 6-Bolt AHC

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A.4 4 Straight Bolts with Crossed Beams and Spot Face

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Figure A-7
4-Straight Bolt AHC

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A.5 Top Hat Concepts

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Figure A-8
Alternate Concept for Flawed Top Hat AHC

(Shroud Support Plate Not Shown)

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Figure A-9
Alternate Concept for Flawed Top Hat AHC

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