WCAP-17894-NP
Revision 0

September 2014

Component Inspection Details Supporting Aging Management of Reactor Internals at Indian Point Unit 2

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September 2014

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AMP	Aging Management Program Plan
AMR	Aging Management Review
ASME	American Society of Mechanical Engineers
B&PV	boiler and pressure vessel
B&W	Babcock & Wilcox
BMI	bottom-mounted instrumentation
BWR	boiling water reactor
CE	Combustion Engineering
CFR	Code of Federal Regulations
CID	Component Inspection Detail
EPRI	Electric Power Research Institute
EVT	enhanced visual testing (a visual NDE method that includes EVT-1)
FMECA	failure mode, effects, and criticality analysis
GALL	Generic Aging Lessons Learned
I&E	Inspection and Evaluation
IASCC	irradiation assisted stress corrosion cracking
ID	identification
INPO	Institute of Nuclear Power Operations
IP2	Indian Point Nuclear Generating Station Unit 2
ISI	in-service inspection
LCP	lower core plate
LRA	License Renewal Application
MRP	materials reliability program
MSC	PWROG Materials Subcommittee
NDE	nondestructive examination
NEI	Nuclear Energy Institute
NRC	Nuclear Regulatory Commission
NSSS	nuclear steam supply system
OEM	Original Equipment Manufacturer
OER	Operating Experience Report
PWR	pressurized water reactor
PWROG	Pressurized Water Reactor Owners Group (formerly WOG)
RCS	reactor coolant system
RVI	reactor vessel internals
SCC	stress corrosion cracking
SER	Safety Evaluation Report
SRP	Standard Review Plan
SSC	systems, structures, and components
UCP	upper core plate
USP	upper support plate
UT	ultrasonic testing (a volumetric NDE method)
VT	visual testing (a visual NDE method that includes VT-1 and VT-3)
WOG	Westinghouse Owners Group

ACKNOWLEDGEMENTS

The authors would like to thank the members of the Entergy Reactor Internals Aging Management Program Team, including Bob Dolansky and our associates at Westinghouse for their efforts in supporting the development of this report.

1 PURPOSE

The management of aging degradation effects in reactor internals is required for nuclear plants considering or entering license renewal, as specified in the United States Nuclear Regulatory Commission (NRC) Standard Review Plan (SRP) [1]. Indian Point Nuclear Generating Station Unit 2 (IP2) will be granted an extended license by the NRC through a Safety Evaluation Report (SER) as documented in NUREG-1930 [2 and 16]. License Renewal Commitment 30 of [2 and 16], "PWR Vessel Internals Program," committed IP2 to:

1. Participate in industry programs for investigating and managing aging effects on reactor internals.
2. Evaluate and implement the results of the industry programs as applicable to the reactor internals.

Upon completion of these programs IP2 submitted an inspection plan for reactor internals to the NRC for review and approval. On September 29, 2013, IP2 entered what is called the period of "timely renewal" while the NRC continues its consideration of Entergy's application to renew its operating license.

An Aging Management Program Plan (AMP) [3] was developed by IP2 that captured the industry guidance for additional reactor internals inspections, based on the programs sponsored by U.S. utilities through the Electric Power Research Institute (EPRI) materials reliability program (MRP) and the Pressurized Water Reactor Owners Group (PWROG). The IP2 AMP was later supplemented by [5] to provide compliance with MRP-227-A.

Additional reactor internals inspections of "Primary" and "Expansion" components per the IP2 AMP are supplemental to the "Existing" RVI component inspections as required per the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel (B&PV) Code, Section XI program. This document contains component inspection details (CIDs) and supplemental information that provides direction on the location, number, configuration, and inspection requirements of the items that comprise the scope for the additional reactor internals inspections at IP2. As such, this document supports the plant-specific implementation of additional reactor internals inspections during the license renewal term.

2 BACKGROUND

There are five key drivers that affect the continued successful operation of reactor vessel internals (RVI) at operating PWRs:

- License Renewal/Life Extension
- Reliability and Maintenance
- Inspection Indications
- Issue-Specific Research
- Component-Specific Aging Management and Inspection Cost Reduction

The Code of Federal Regulations (CFR) Maintenance Rule, 10 CFR 50.65 requires monitoring of certain systems, structures, and components (SSCs) against established goals to provide reasonable assurance that those SSCs are capable of fulfilling their intended functions. License renewal requirements of 10 CFR Part 54, require a plant to demonstrate effective management of aging, shifting the emphasis from identifying aging mechanisms to managing their effects during the license renewal period. Together these two regulatory requirements consider all active and passive components that are required for safe operation of the plant with aging management focusing on passive, long-lived structures and components. A plant entering or requesting license renewal is typically required to define and execute an AMP for reactor internals.

For many years the U.S. nuclear power industry has been actively engaged in efforts to support the industry goal of responding to the regulatory requirements on managing aging degradation in reactor internals. Various programs have been underway to develop guidelines for managing the effects of aging within PWR reactor internals. Westinghouse Owners Group (WOG) WCAP-14577 [6] received NRC Staff review and approval and served as the initial basis for developing AMPs for RVIs. The industry efforts to address the concern were continued by the EPRI MRP and included consideration of the three currently operating U.S. reactor designs – Westinghouse, Combustion Engineering (CE), and Babcock & Wilcox (B&W).

The MRP established a framework and strategy for the aging management of PWR internals components using proven and familiar methods for inspection, monitoring, surveillance, and communication. Factoring in the accumulated industry research data, the following elements of an AMP were further refined [7 and 8]:

- Screening criteria were developed, considering chemical composition, neutron fluence exposure, temperature history, and representative stress levels, for determining the relative susceptibility of PWR internals components to each of eight postulated aging mechanisms.

- PWR internals components were categorized, based on the screening criteria, as follows:

  - Components for which the effects from the postulated aging mechanisms are insignificant

  - Components that are moderately susceptible to the aging effects

  - Components that are significantly susceptible to the aging effects

- Functionality assessments were performed based on representative PWR internals components and component assemblies using irradiated and aged material properties, to determine the effects of the degradation mechanisms on component functionality.

Aging management strategies were developed combining the functionality assessment results with contributing factors to determine the appropriate aging management methodology, baseline examination timing, and the need and timing of subsequent inspections. Items considered included component accessibility, operating experience, existing evaluations, and prior examination results.

The industry finalized initial Inspection and Evaluation (I&E) Guidelines for reactor internals and submitted the document to the NRC with a request for a formal SER. A supporting document addressing inspection requirements was also completed and provided to the NRC to support the I&E Guidelines review. A third document, which was generated by the industry through the PWROG Materials Subcommittee (MSC), provides detailed engineering criteria for evaluating acceptance of inspection outcomes. This industry developed guidance is contained within the following three documents:

- MRP-227-A [9] (hereafter referred to as “the I&E Guidelines” or simply “MRP-227-A”) provides the industry background, listing of generic reactor internals components requiring inspection, type of nondestructive examination (NDE) required for each component, timing for initial inspections, and direction for evaluating inspection results.

- MRP-228 [10] provides guidance on the qualification and demonstration of the NDE techniques and other criteria pertaining to the actual performance of the inspections.

- WCAP-17096-NP [11] provides direction on engineering evaluations of inspection outcomes to determine acceptability for continued service.

The IP2 reactor internals are integral with the reactor coolant system (RCS) of a Westinghouse four-loop nuclear steam supply system (NSSS). IP2 reactor internals have a downflow baffle-barrel region flow design, and a top hat design upper support plate (USP). An illustration of IP2 internals is provided in Figure A-2.

As described in NUREG-1930 [2 and 16], the applicant described the RVIs, which consist of two basic assemblies: an upper internals assembly that is removed during each refueling operation to obtain access to the reactor core and a lower internals assembly that can be removed following a complete core off-load.

The reactor core is positioned and supported by the upper internals and lower internals assemblies. The individual fuel assemblies are positioned by fuel alignment pins in the upper core plate (UCP) and the lower core plate (LCP). These pins control the orientation of the core with respect to the upper and lower internals assemblies. The lower internals are aligned with the upper internals by the UCP alignment pins and secondarily by the head/vessel alignment pins. The lower internals are aligned to the vessel by the lower radial support/clevis assemblies and by the head/vessel alignment pins. Thus, the core is aligned with the vessel by a number of interfacing components.

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

Upper Internals Assembly

The major sub-assemblies that constitute the upper internals assembly are the: (1) UCP; (2) upper support column assemblies; (3) control rod guide tube assemblies; and (4) USP.

During reactor operation, the upper internals assembly is preloaded against the fuel assembly springs and the internals hold down spring by the reactor vessel head pressing down on the outside edge of the USP. The upper support columns and the control rod guide tubes are attached to the USP. The UCP, in turn, is attached to the upper support columns. The USP design at IP2 is designated as a top hat design.

The UCP is perforated to permit coolant to pass from the core below into the upper plenum defined by the USP and UCP. The UCP positions and laterally supports the core by fuel alignment pins extending below the plate. The UCP contacts and preloads the fuel assembly springs and thus maintains contact of the fuel assemblies with the LCP during reactor operation.

The upper support columns vertically position the UCP and are designed to take the uplifting hydraulic flow loads and fuel spring loads on the UCP. The control rod guide tubes are bolted to the USP and pinned at the UCP. Guide tube cards are located within the control rod guide tube assembly to guide the absorber rods. The control rod guide tubes are also slotted in their lower sections to allow coolant exiting the core to flow into the upper plenum.

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

Lower Internals Assembly

The fuel assemblies are supported inside the lower internals assembly on top of the LCP. The functions of the LCP are to position and support the core and provide a metered control of reactor coolant flow into each fuel assembly. The LCP is elevated above the lower support casting by support columns and bolted to a ring support attached to the inside diameter of the core barrel. The support columns transmit vertical fuel assembly loads from the LCP to the much thicker lower support casting, which provides support for the core.

The primary function of the core barrel is to support the core. A large number of components are attached to the core barrel, including the baffle/former assembly, the core barrel outlet nozzles, the thermal shield, the alignment pins that engage the UCP, the lower support casting, and the LCP. The lower radial support/clevis assemblies restrain large transverse motions of the core barrel, but at the same time allow unrestricted radial and axial thermal expansion.

The baffle and former assembly consists of vertical plates called baffles and horizontal support plates called formers. The baffle plates are bolted to the formers by the baffle/former bolts, and the formers are attached to the core barrel inside diameter by the barrel/former bolts. Baffle plates are secured to each other at selected corners by edge bolts. In addition, at IP2, corner brackets are installed behind and bolted to the baffle plates. The baffle/former assembly forms the interface between the core and the core barrel. The baffles provide a barrier between the core and the former region so that a high concentration of flow in the core region can be maintained. A secondary benefit is to reduce the neutron flux on the vessel. Additional neutron shielding of the reactor vessel is provided in the active core region by the thermal shield attached to the outside of the core barrel.

In the upper internals assembly, the USP, the upper support columns, and the UCP are considered core support structures. In the lower internals assembly, the LCP, the lower support casting, the lower support columns, the core barrel including the core barrel flange, the radial support/clevis assemblies, the baffle plates, and the former plates are classified as core support structures.

All RVIs are removable for their inspection, and for inspection of the vessel internal surface.

Based on the completed evaluations, the RVI components are categorized within MRP-227-A as “Primary” components, “Expansion” components, “Existing Programs” components, or “No Additional Measures” components. Descriptions of the final categories are as follows:

- Primary

  Those PWR internals that are highly susceptible to the effects of at least one of the eight aging mechanisms were placed in the Primary group. The aging management requirements that are needed to ensure functionality of Primary components are described in the I&E guidelines. The Primary group also includes components that have shown a degree of tolerance to a specific aging degradation effect, but for which no highly susceptible component exists or for which no highly susceptible component is accessible.

- Expansion

  Those PWR internals that are highly or moderately susceptible to the effects of at least one of the eight aging mechanisms, but for which functionality assessment has shown a degree of tolerance to those effects, were placed in the Expansion group. The schedule for implementation of aging management requirements for Expansion components depends on the findings from Primary component examinations at individual plants.

- Existing Programs

  Those PWR internals that are susceptible to the effects of at least one of the eight aging mechanisms and for which generic and plant-specific existing AMP elements are capable of managing those effects, were placed in the Existing Programs group.

- No Additional Measures Programs

  Those PWR internals for which the effects of all eight aging mechanisms are below the screening criteria were placed in the No Additional Measures group. Additional components were placed in the No Additional Measures group as a result of a failure mode, effects, and criticality analysis (FMECA) and the functionality assessment. No further action is required by these guidelines for No Additional Measures components aging management.

The categorization and analysis used in the development of MRP-227-A were not intended to supersede any ASME B&PV Code Section XI requirements. Components that are classified as core support structures, as defined in ASME B&PV Code Section XI IWB 2500, Category B-N-3, have requirements that remain in effect and may only be altered as allowed by 10 CFR 50.55a.

A listing of the IP2 RVI components and subcomponents already reviewed by the NRC in the SER granting life extension that are subject to aging management requirements were included as Tables 5-2, 5-3, and 5-4 of the IP2 License Renewal Application [5]. The link between primary and expansion MRP-227-A components is defined in Table 5-5 of [5]. The IP2-specific MRP-227-A reactor internals components that require additional inspections and the components with existing ASME Section XI inspections that are credited for managing aging in RVI are summarized by MRP-227-A classification categories and shown in Table 4-1 of this report.

3 PROGRAM OWNER

The PWR Vessel Internals Program is established in accordance with Entergy's "NEI 03-08 Materials Initiative Process" [12]. The successful implementation and comprehensive long-term management of the IP2 RVI AMP will require the integration of Entergy organizations (both corporate and at IP2) and interaction with multiple industry organizations including, but not limited to, the ASME, MRP, NRC, and PWROG. The responsibilities of the individual corporate and IP2 groups are delineated in appropriate site procedures. Entergy will maintain cognizance of industry activities related to PWR internals inspection and aging management, and will address/implement industry guidance stemming from those activities, as appropriate under Nuclear Energy Institute (NEI) NEI 03-08 [13] practices. The appropriate personnel and their responsibilities are summarized in site procedures.

4 COMPONENT INSPECTION DETAILS OF THE INDIAN POINT UNIT 2 REACTOR INTERNALS

4.1 INTRODUCTION

The U.S. nuclear industry, through the combined efforts of utilities, vendors, and independent consultants, defined a generic guideline to assist utilities in developing reactor internals plant-specific aging management programs based on inspection and evaluation. The primary objective for the industry effort and each individual plant program is to ensure the long-term integrity and safe operation of the reactor internals components and overall reliability of the plant by proactively managing aging.

The IP2 reactor internals AMP utilizes a combination of prevention, mitigation, and condition monitoring to manage aging in susceptible reactor internals components. Where applicable, credit is taken for existing programs such as water chemistry [7] and inspections prescribed by the ASME Section XI In-Service Inspection (ISI) Program [4], combined with additional reactor internals inspections or evaluations as recommended by MRP-227-A.

The purpose of this IP2 CID WCAP is to support IP2-specific implementation of the generic industry additional reactor internals inspections with a focus on components that are not already included in existing IP2 inspection programs. To ensure a clear understanding of the requirements, a brief description of the required inspections is included in this section followed by descriptions of the individual components that comprise the IP2 additional reactor internals MRP-227-A inspections. CIDs illustrating key considerations of the required MRP-227-A inspection for the component is included for all applicable IP2 MRP-227-A Primary and Expansion components

A listing of the components applicable to IP2 identifying the MRP-227-A category, required inspection, and corresponding CID is provided in Table 4-1. The Control Rod Guide Tube Assembly: Guide Tube Support Pins (Split Pins) are not included in Table 4-1 because they are managed by Original Equipment Manufacturer (OEM) recommendations in accordance with Applicant/Licensee Action Item 3 of the NRC Safety Evaluation for MRP-227 as described in Section 3.6 of NL-12-037 [5].

Table 4-1 IP2 Reactor Internals Components Requiring Additional Inspections During the License Renewal Term

Component	MRP-227-A		CID	WCAP-17096-NP ID(2)
	Category	Inspection Type		
Control Rod Guide Tube Assembly - Guide Plates (Cards)	Primary	VT-3	IP2-CID-0010	W-ID: 1
Control Rod Guide Tube Assembly - Lower Flange Welds	Primary	EVT-1	IP2-CID-0020	W-ID: 2
Upper Internals Assembly - Upper Core Plate	Expansion	EVT-1	IP2-CID-0021	W-ID: 2.1
Lower Internals Assembly - Lower Support Forging or Castings(3)	Expansion	EVT-1	IP2-CID-0022	W-ID: 2.2
Lower Support Assembly - Lower Support Column Bodies (Cast)	Expansion	EVT-1	IP2-CID-0023	W-ID: 2.3
Bottom-mounted Instrumentation System - Bottom-mounted Instrumentation (BMI) Column Bodies	Expansion	VT-3	IP2-CID-0024	W-ID: 2.4
Core Barrel Assembly - Upper Core Barrel Flange Weld	Primary	EVT-1	IP2-CID-0030	W-ID: 3
Core Barrel Assembly - Core Barrel Outlet Nozzle Welds	Expansion	EVT-1	IP2-CID-0031	W-ID: 3.1
Core Barrel Assembly - Upper and Lower Core Barrel Cylinder Girth Welds	Primary	EVT-1	IP2-CID-0040	W-ID: 4
Core Barrel Assembly - Upper and Lower Core Barrel Cylinder Axial Welds	Expansion	EVT-1	IP2-CID-0041	W-ID: 4.1
Core Barrel Assembly - Lower Core Barrel Flange Weld(4)	Primary	EVT-1	IP2-CID-0050	W-ID: 5
Baffle-former Assembly - Baffle-edge Bolts	Primary	VT-3	IP2-CID-0060	W-ID: 6
Baffle-former Assembly - Baffle-former Bolts	Primary	UT	IP2-CID-0070	W-ID: 7
Core Barrel Assembly - Barrel-former Bolts	Expansion	UT	IP2-CID-0071	W-ID: 7.1
Lower Support Assembly - Lower Support Column Bolts	Expansion	UT	IP2-CID-0072	W-ID: 7.2
Baffle-former Assembly - Baffle-former Assembly	Primary	VT-3	IP2-CID-0080	W-ID: 8
Alignment and Interfacing Components – Internals Hold-down Spring	Primary	Special(1)	IP2-CID-0090	W-ID: 9
Thermal Shield Assembly - Thermal Shield Flexures	Primary	VT-3	IP2-CID-0100	W-ID: 10
Core Barrel Assembly: Core Barrel Flange	Existing	VT-3	Not applicable	Not applicable
Upper Internals Assembly: Upper Support Ring or Skirt(5)	Existing	VT-3	Not applicable	Not applicable

Table 4-1 IP2 Reactor Internals Components Requiring Additional Inspections During the License Renewal Term (cont.)

Component	MRP-227-A		CID	WCAP-17096-NP ID(2)
	Category	Inspection Type		
Lower Core Plate	Existing	VT-3	Not applicable	Not applicable
Alignment and Interfacing Equipment: Clevis Insert Bolts	Existing	VT-3	Not applicable	Not applicable
Alignment and Interfacing Equipment: Upper Core Plate Alignment Pins	Existing	VT-3	Not applicable	Not applicable
Bottom Mounted Instrumentation System: Flux Thimble Tubes	Existing	ET	Not applicable	Not applicable

Notes:

1. Managed by plant-specific measurement programs.
2. Confirmation of identification (ID) number pending NRC approval of WCAP-17096-NP.
3. This component is a casting at IP2.
4. At IP2 this weld is the lower core barrel to lower support casting weld. IP2 does not have a lower core barrel flange.
5. IP2 has a tophat design. Therefore, there is no support ring or skirt; however, the vertical sections of the tophat will be inspected.

4.2 DETECTION OF AGING EFFECTS

Inspection can be used to detect physical effects of degradation including cracking, fracture, wear, and distortion. The inspection technique is chosen based on the nature and extent of the expected damage. The recommendations supporting aging management for the reactor internals, as defined by the industry and contained in this report, are built around three basic inspection techniques: (1) visual testing (VT), (2) ultrasonic testing (UT), and (3) physical measurement. Three different visual techniques are included: VT-3, VT-1, and EVT-1. Those additional reactor internals inspections that are taken from the MRP-227-A recommendations will be applied through use of the MRP-228 Inspection Standard [10]. Detection of indications that are required by the ASME Section XI ISI Program is well-established and field-proven through the application of the Section XI ISI Program.

The assumptions and process used to select the appropriate inspection technique are described in the following subsections. Inspection standards developed by the industry for the application of these techniques for additional reactor internals inspections are documented in MRP-228.

VT-1 Visual Examinations

In MRP-227-A note that VT-1 has only been selected to detect distortion as evidenced by small gaps between the upper-to-lower mating surfaces of CE-welded core shrouds assembled in two vertical sections. Therefore, no additional VT-1 inspections over and above those required by ASME Section XI ISI have been specified in MRP-227-A at IP2.

EVT-1 Enhanced Visual Examination for the Detection of Surface Breaking Flaws

In the additional reactor internals inspections detailed in the MRP-227-A, the EVT-1 enhanced visual examination has been identified for inspection of components where surface-breaking flaws are a potential concern. Any visual inspection for cracking requires a reasonable expectation that the flaw length and crack mouth opening displacement meet the resolution requirements of the observation technique. The EVT-1 specification augments the VT-1 requirements to provide more rigorous inspection standards for stress corrosion cracking (SCC) and has been demonstrated for similar inspections in boiling water reactor (BWR) internals. Enhanced visual examination (i.e., EVT-1) is also conducted in accordance with the requirements described for visual examination (i.e., VT-1) with additional requirements (such as camera scanning speed) currently being developed by the industry. Any recommendation for EVT-1 inspection will require additional analysis to establish flaw-tolerance criteria. The industry is currently developing a consensus approach for acceptance criteria methodologies to support plant-specific additional reactor internals examinations. Entergy has been an active participant in these initiatives and will follow the industry directive. These acceptance criteria methodologies may be determined either generically or on a plant-specific basis because both loads and component dimensions may vary from plant to plant within a typical PWR design.

VT-3 Examination for General Condition Monitoring

In the additional reactor internals inspections detailed in the MRP-227-A, the VT-3 visual examination has been identified for inspection of components where general condition monitoring is required. The VT-3 examination is intended to identify individual components with significant levels of existing

degradation. As the VT-3 examination is not intended to detect the early stages of component cracking or other incipient degradation effects, it should not be used when failure of an individual component could threaten either plant safety or operational stability. The VT-3 examination may be appropriate for inspecting highly redundant components (such as baffle-edge bolts), where a single failure does not compromise the function or integrity of the critical assembly.

The acceptance criteria for visual examinations conducted under categories B-N-2 (welded core support structures and interior attachments to reactor vessels) and B-N-3 (removable core support structures) are defined in IWB-3520 [14]. These criteria are designed to provide general guidelines. The unacceptable conditions for a VT-3 examination are:

- Structural distortion or displacement of parts to the extent that component function may be impaired
- Loose, missing, cracked, or fractured parts, bolting, or fasteners
- Foreign materials or accumulation of corrosion products that could interfere with control rod motion or could result in blockage of coolant flow through fuel
- Corrosion or erosion that reduces the nominal section thickness by more than five percent
- Wear of mating surfaces that may lead to loss of function
- Structural degradation of interior attachments such that the original cross-sectional area is reduced more than five percent

The VT-3 examination is intended for use in situations where the degradation is readily observable. It is meant to provide an indication of condition, and quantitative acceptance criteria are not generally required. In any particular recommendation for VT-3 visual examination, it should be possible to identify the specific conditions of concern. For instance, the unacceptable conditions for wear indicate wear that might lead to loss of function. Guidelines for wear in a critical-alignment component may be very different from the guidelines for wear in a large structural component.

<u>Ultrasonic Testing</u>

Volumetric examinations in the form of UT techniques can be used to identify and determine the length and depth of a crack in a component. Although access to the surface of the component is required to apply the ultrasonic signals, the flaw may exist in the bulk of the material. In this proposed strategy, UT inspections have been recommended exclusively for detection of flaws in bolts. For the bolt inspections, any bolt with a detected flaw should be assumed to have failed. The size of the flaw in the bolt is not critical because crack growth rates are generally high, and it is assumed that the observed flaw will result in failure prior to the next inspection opportunity. It has generally been observed through examination performance demonstrations that UT can reliably (90 percent or greater reliability) detect flaws that reduce the cross-sectional area of a bolt by 35 percent.

Failure of a single bolt does not compromise the function of the entire assembly. Bolting systems in the reactor internals are highly redundant. For any system of bolts, it is possible to demonstrate multiple minimum acceptable bolting patterns. The evaluation program must demonstrate that the remaining bolts meet the requirements for a minimum bolting pattern for continued operation. The evaluation procedures must also demonstrate that the pattern of remaining bolts contains sufficient margin such that continuation of the bolt failure rate will not result in failure of the system to meet the requirements for minimum acceptable bolting pattern before the next inspection.

Establishment of the minimum acceptable bolting pattern for any system of bolts requires analysis to demonstrate that the system will maintain reliability and integrity in continuing to perform the intended function of the component. This analysis is highly plant-specific. Therefore, any recommendation for UT inspection of bolts assumes that the plant owner will work with the designer to establish minimum acceptable bolting patterns prior to the inspection to support continued operation. For Westinghouse-designed plants, minimum acceptable bolting patterns for baffle-former and barrel-former bolts are available through the PWROG. Entergy has been a full participant in the development of the PWROG documents and has full access and use.

Physical Measurement Examination

Continued functionality can be confirmed by physical measurements to evaluate the impact caused by various degradation mechanisms such as wear or loss of functionality as a result of loss of preload or material deformation. For IP2, direct physical measurements are required only for the internals hold-down spring. An alternate option is to replace the existing internals hold-down spring which could eliminate the need for measurements.

4.3 INSPECTION RESULTS REPORTING FORMAT

Entergy IP2, the MRP, and the PWROG approaches to aging management are based on the Generic Aging Lessons Learned (GALL)Report approach as detailed in NUREG-1801 [15]. This approach includes determining which reactor internals passive components are most susceptible to the aging mechanisms of concern and then determining the proper inspection or mitigating program that provides reasonable assurance that the component will continue to perform its intended function through the period of extended operation. The GALL-based approach was used at IP2 for the initial basis of the License Renewal Application (LRA) that resulted in the NRC SER in NUREG-1930 [2 and 16].

A key element of the MRP-227-A guideline is the reporting of age-related degradation of reactor vessel components through Operating Experience Reports (OERs). Entergy, through its participation in PWROG and EPRI-MRP activities, will continue to benefit from the reporting of inspection information and will share its own operating experience with the industry through those groups or Institute of Nuclear Power Operations (INPO), as appropriate.

The component naming nomenclature implemented by the industry in PWROG WCAP-17096 [11] forms the basis for data recording of inspection results. WCAP-17096-NP is currently being updated to account for changes from MRP-227 Rev. 0 to MRP-227-A, which is why the W-ID numbers are listed as “pending”. Location indicators for each component item are based on the IP2-specific design.

4.4 COMPONENTS

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

All Westinghouse internals consist of two basic assemblies: an upper internals assembly that is removed during each refueling operation to obtain access to the reactor core, and a lower internals assembly that can be removed following a complete core off-load. The lower internals assembly is supported in the vessel by clamping to a ledge below the vessel head mating surface and is closely guided at the bottom by radial support clevis assemblies. The upper internals assembly is clamped at this same ledge by the reactor vessel head. The bottom of the upper internals assembly is closely guided by the core barrel alignment pins of the lower internals assembly.

All of the assemblies and individual components making up the Westinghouse reactor internals were considered in the process of developing the MRP-227-A requirements. The complete categorization process is summarized in MRP-227-A and supporting basis documents. Brief descriptions of the components required to support aging management of reactor internals at IP2 are included in the following subsections by MRP-227-A Primary, Expansion, and Existing categories.

4.4.1 Primary

MRP-227-A Primary components are those that were determined to be highly susceptible to the effects of a least one of the critical degradation mechanisms affecting reactor internals or a component for which no highly susceptible component exists or was directly accessible. Table 4-1 contains a complete listing of all of the Primary components applicable to IP2.

Control Rod Guide Tube Assembly - Guide Plates (Cards)

The control rod guide tube assembly provides an alignment and insertion path for the control rods through the upper internals. Guide cards provide alignment and an insertion path for control rod assemblies, and support the control rods when withdrawn. The guidance holes in the guide cards are distorted by wear (loss of material). The largest amounts of wear to date have typically been observed in the lowest guide card levels.

Guidance hole wear can cause lack of alignment. Lack of alignment may impact control rod drop times. In the worst-case scenario, control rods may jam and prevent full insertion which would cause a technical specification compliance concern.

WCAP-17096-NP Component Category ID: W-ID: 1 (pending)

Control Rod Guide Tube Assembly - Lower Flange Welds

The control rod guide tube assembly provides alignment and an insertion path for control rods through the upper internals. The lower flange welds retain the structural alignment and stability of the component.

Flow in the upper head applies a bending moment to the control rod guide tube assembly. Maximum bending stresses tend to occur near the top of the continuous guidance section (upper flange weld location as identified in Figure A-4). Stresses may lead to formation of SCC or fatigue cracks. Weld cracking may lead to loss of stiffness in the guide tube assembly and loss of support capability yielding a loss of structural stability. Excessive deflection could impede control assembly insertion, resulting in a failure to perform its intended function.

WCAP-17096-NP Component Category ID: W-ID: 2 (pending)

Core Barrel Assembly - Upper Core Barrel Flange Weld

The upper core barrel flange weld is an integral component of the primary core support structure. The primary concern with regard to aging is through-wall cracking in the weld as a result of stress corrosion. Actively growing through-wall flaws would require attention and could cause a potential loss of core support resulting in safety concerns. However, the core barrel is considered a highly flaw-tolerant structure, and relatively large inactive flaws are likely to be manageable even if found.

WCAP-17096-NP Component Category ID: W-ID: 3 (pending)

Core Barrel Assembly – Upper and Lower Core Barrel Cylinder Girth Welds

The core barrel cylinder girth welds are an integral component of the primary core support structure. The primary concern with regard to aging is through-wall cracking in the weld as a result of stress corrosion. Actively growing through-wall flaws would require attention and could cause a potential loss of core support resulting in safety concerns. However, the core barrel is considered a highly flaw-tolerant structure, and relatively large inactive flaws are likely to be manageable even if found.

WCAP-17096-NP Component Category ID: W-ID: 4 (pending)

Core Barrel Assembly – Lower Flange Weld

The lower flange weld is an integral component of the primary core support structure. This weld joins the core barrel cylinder to the lower core support forging/casting; no actual flange exists at this location. The primary concern with regard to aging is through-wall cracking in the weld as a result of stress corrosion. Actively growing through-wall flaws would require attention and could cause a potential loss of core support resulting in safety concerns. However, the core barrel is considered a highly flaw tolerant structure and relatively large inactive flaws are likely to be manageable even if found.

WCAP-17096-NP Component Category ID: W-ID: 5 (pending)

Baffle-former Assembly - Baffle-edge Bolts

The baffle-edge bolts provide baffle-plate to baffle-plate attachment along the seams between the plates and function to prevent gaps forming between plates. Structural studies have demonstrated that baffle-edge bolts are not required to maintain the structural integrity of the baffle; therefore, baffle-edge bolts are not considered to be a safety significant component. Analysis has shown that differential thermal expansion and swelling can cause plastic deformation of edge bolts. The edge bolts are in high radiation locations, and a significant potential for failure exists due to irradiation assisted stress corrosion cracking (IASCC). Operationally, gaps between plates can result in baffle-jetting damage to fuel assemblies. In plants with downward coolant flow in the region between the baffle and the former, failure of baffle-edge bolts is considered to directly contribute to baffle jetting.

Evidence of baffle-edge bolt failure would typically be observed through broken or missing locking devices, protruding bolt heads, or missing bolts or bolt heads. Therefore, the primary concerns are baffle jetting, loose parts generation, and interference with fuel from broken or missing baffle-edge bolts.

WCAP-17096-NP Component Category ID: W-ID: 6 (pending)

Baffle-former Assembly - Baffle-former Bolts

The baffle-former bolts attach the baffle plates to the baffle formers. Documented observations of IASCC cracking of these components exists in multiple designs in the PWR fleet worldwide. These highly irradiated bolts perform a critical safety and operational function in the plant. Loss of a single bolt or isolated multiple failures of the baffle-former bolts are considered to be manageable, but a catastrophic or

clustered loss of multiple bolts at adjacent locations could cause a lack of structural stability and potentially raise safety and operational concerns.

WCAP-17096-NP Component Category ID: W-ID: 7 (pending)

Baffle-former Assembly – Assembly

The baffle-former assembly is made up of vertical plates named "baffles" and horizontal support plates named "formers." The baffle plates are bolted to the formers by the baffle-former bolts, and the formers are attached to the core barrel inside surface by the barrel-former bolts. Some of the baffle plates are also bolted to each other at selected corners by edge bolts or brackets. The baffle-former assembly forms the interface between the core and the core barrel.

The baffle-former assembly provides support, guidance, and protection for the reactor core, a passageway for the distribution of the reactor coolant flow to the reactor core, and radiation (gamma and neutron) shielding for the reactor vessel. Void swelling and IASCC of the integrated assembly are the key concerns identified which could manifest multiple degradation effects, such as interference with fuel assemblies, obstruction of coolant flow, loose parts generation, distortion and misalignment of the core, local temperature peaks, degradation of control rod insertion paths, and baffle jetting. To date, no relevant observations of these effects in the baffle-former assembly as a result of void swelling or IASCC have been documented.

WCAP-17096-NP Component Category ID: W-ID: 8 (pending)

Alignment and Interfacing Components – Internals Hold-Down Spring

The function of the hold-down spring is to retain the reactor internals in proper alignment to the core. The primary aging degradation concern is stress relaxation as a result of long-term service. The direct result of long term stress relaxation is a loss of hold-down forces leading to vibration and wear in the lower internals.

WCAP-17096-NP Component Category ID: W-ID: 9 (pending)

Thermal Shield Assembly - Thermal Shield Flexures

Thermal shield flexures are part of the thermal shield support system. This system provides primary attachment of the thermal shield to the core barrel at the lower connection points of the assembly. The thermal shield flexures provide the lower structural support for the thermal shield and are required to hold the bottom of the thermal shield concentric to the core barrel. Due to the differential deflections between the core barrel and thermal shield caused by thermal cycling resulting from plant operation, the thermal shield flexures are potentially susceptible to fatigue. Indications of wear in the components (bolts, pins, and fasteners) composing the thermal shield-to-core barrel attachment system, failure in the welds at the base of the flexure, or failure in the weld that attaches the flexure to the thermal shield are the identified indicators of age-related concerns in the assembly.

WCAP-17096-NP Component Category ID: W-ID: 10 (pending)

4.4.2 Expansion

MRP-227-A Expansion components are those that were determined to be highly or moderately susceptible to the effects of a least one of the critical degradation mechanisms affecting reactor internals but which detailed evaluations showed a degree of tolerance to those effects. Examination of expansion components is dependent on the plant-specific primary component inspection observations and the expansion inspection requirements in Table 4-6 of MRP-227-A [9] and Table 5-3 of NL-12-037 [5].

Table 4-1 lists all of the Expansion components applicable to IP2.

<u>Upper Internals Assembly - Upper Core Plate</u>

The UCP is perforated to permit coolant to pass from the core into the upper plenum defined by the USP and the UCP. The coolant then exits through the outlet nozzles in the core barrel. The UCP positions and laterally supports the core by fuel pins extending below the plate. The UCP contacts and preloads the fuel assembly springs and, therefore, maintains contact of the fuel assemblies with the LCP during reactor operation. Aging concerns for this component are cracking from wear and fatigue, which may compromise the ability of the UCP to properly align the fuel.

WCAP-17096-NP Component Category ID: W-ID: 2.1 (pending)

<u>Lower Internals Assembly - Lower Support Forging or Casting</u>

The lower support forging or casting provides support for the core. Indian Point has a lower support casting. Cracking resulting in displacement of lower support forgings or castings would cause concerns for the operational and structural stability of the lower reactor internals support assembly. The lower support casting is welded to and supported by the core barrel, which transmits vertical loads to the vessel through the core barrel flange. Aging concerns for this component are cracking and thermal embrittlement.

WCAP-17096-NP Component Category ID: W-ID: 2.2 (pending)

<u>Lower Support Assembly - Lower Support Column Bodies (Cast)</u>

The lower support columns provide the structural link between the LCP and the lower support structure. The supports are required to keep the LCP from deforming during plant operation. The upper sections of the lower support column bodies may experience neutron fluences above the industry thresholds for IASCC. The IP2 lower support column bodies are composed of forged and cast materials.

The cast components are considered separately because a concern exists that they may be more sensitive to thermal and irradiation effects. Although stresses in columns are primarily compressive, bending stresses or the design of the attachment may produce localized regions of tensile stress and, therefore, have increased susceptibility to cracking. Maintaining core stability and core plate flatness is the intended purpose of this component, and loss of one or more may compromise this function.

WCAP-17096-NP Component Category ID: W-ID: 2.3 (cast) (pending)

Bottom-mounted Instrumentation System - Bottom-mounted Instrumentation (BMI) Column Bodies

The BMI column bodies define the path for flux thimbles to be inserted into the fuel assemblies. The plant must maintain a required number of functioning flux thimbles for core mapping. Flux thimbles are normally withdrawn prior to refueling and reinserted at the end of the refueling activities. A key consideration is that the primary pressure boundary must remain intact.

The BMI column bodies may be subject to fatigue due to either thermal fatigue or flow-induced vibrations. Inability to insert flux thimbles would be noted during refueling activities. Once flux thimbles are inserted, the consequences of failure of the component to perform its intended function during the ensuing operating cycle are typically considered to be minimal.

WCAP-17096-NP Component Category ID: W-ID: 2.4 (pending)

Core Barrel Assembly - Core Barrel Outlet Nozzle Welds

The core barrel outlet nozzle welds join the core barrel outlet nozzles to the upper core barrel cylinder. The primary concern with regard to aging is through-wall cracking in the weld as a result of stress corrosion. Actively growing through-wall flaws would require attention and could potentially cause jetting through the core barrel. However, the core barrel is considered a highly flaw-tolerant structure, and relatively large inactive flaws are likely to be manageable even if found.

WCAP-17096-NP Component Category ID: W-ID: 3.1 (pending)

Core Barrel Assembly - Upper and Lower Core Barrel Cylinder Axial Welds

The core barrel axial welds are in place to maintain the cylindrical shape of the core barrel, as the core barrel cylinders originate as flat plates that are then rolled into cylinders. The primary concern with regard to aging is through-wall cracking in the weld as a result of stress corrosion. Actively growing through-wall flaws would require attention and could potentially cause jetting through the core barrel. However, the core barrel is considered a highly flaw-tolerant structure, and relatively large inactive flaws are likely to be manageable even if found.

WCAP-17096-NP Component Category ID: W-ID: 4.1 (pending)

Core Barrel Assembly - Barrel-former Bolts

The barrel-former bolts join the barrel to the former plates and are vital to maintaining the operational and structural integrity of the integrated baffle-former-barrel assembly. The primary concern is bolt cracking due to IASSC and fatigue. A loss of bolt preload due to irradiation-induced stress relaxation may exacerbate fatigue issues in aging plants. The potential for flow-induced vibration due to loss of bolting constraint would contribute to overall loss of function. Loss of structural stability is an operational and safety concern.

WCAP-17096-NP Component Category ID: W-ID: 7.1 (pending)

Lower Support Assembly - Lower Support Column Bolts

The lower support column bolts attach the support columns to the lower core support plate. Although the bolts do not directly support the weight of the core, they help maintain the flatness and integrity of the lower core support plate. Cracking from IASCC or fatigue resulting in displacement of the lower core support plate would cause concerns for the operational and structural stability of the lower reactor internals support assembly.

WCAP-17096-NP Component Category ID: W-ID: 7.2 (pending)

4.4.3 Existing

MRP-227-A Existing components are those that were determined to be susceptible to the effects of a least one of the critical degradation mechanisms affecting reactor internals, but for which existing plant-specific AMP elements were found to be sufficient to manage aging concerns. The MRP-227-A requirement is for IP2 to ensure that the MRP-227-A [9], Table 4-9, and NL-12-037 [5], Table 5-4, exams are included in plant-specific inspection programs.

Table 4-1 lists all of the Existing components applicable to IP2.

5 CONCLUSION

Indian Point Nuclear Generating Station Unit 2 has demonstrated a long-term commitment to aging management of reactor internals. The additional evaluations and analyses completed by the MRP industry group have provided clarification to the level of inspection quality needed to determine the proper examination method and frequencies. It is the industry position that use of the Aging Management Review (AMR) produced by the LRA methodology, combined with any additional reactor internals inspections required by the MRP-227-A industry tables provided in MRP-227-A and the plant-specific AMP, provides reasonable assurance that the reactor internals passive components will continue to perform their intended functions through the period of extended operation.

ASME Section XI examinations identified in the AMP for the period of extended operation and additional reactor internals inspections discussed in compliance with MRP-227-A requirements as an integrated inspection program aligned with ASME Section XI 10-year ISI examinations will be tracked by plant-specific procedures and programs. As discussed, the industry MRP-227-A guidelines also provide for updates as experience is gained through inspection results. This feedback loop will enable updates based on actual inspection experience.

The additional reactor internals inspections described in this document, combined with the ASME Section XI ISI program inspections, existing IP2 programs, and use of OERs provide reasonable assurance that the reactor internals will continue to perform their intended functions through the period of extended operation and are in full compliance with IP2 commitments to manage material degradation in reactor internals.

6 REFERENCES

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2. U.S. Nuclear Regulatory Commission NUREG-1930, Vols. 1 and 2, "Safety Evaluation Report Related to the License Renewal of Indian Point Nuclear Generating Unit Nos. 2 and 3," November 2009.
3. Entergy Nuclear Engineering Report, IP-RPT-11-00036, Rev. 0, "Indian Point Energy Center Reactor Vessel Internals Program," October 3, 2011.
4. Entergy Document, SEP-ISI-IP2-001, Rev. 2, "IP2 Fourth Ten-Year Interval Inservice Inspection (ISI)/Containment Inservice Inspection (CII) Program Plan, September 12, 2013.
5. Entergy Letter, NL-12-037, Rev. 0, "License Renewal Application – Revised Reactor Vessel Internals Program and Inspection Plan Compliant with MRP-227-A, Indian Point Nuclear Generation Unit Nos. 2 and 3, Docket Nos. 50-247 and 50-286, License Nos. DPR-26 and DPR-64," February 17, 2012.
6. Westinghouse Report, WCAP-14577, Rev. 1-A, "License Renewal Evaluation: Aging Management for Reactor Internals," March 2001.
7. *Pressurized Water Reactor Primary Water Chemistry Guidelines, Volumes 1 and 2, Revision 6*. EPRI, Palo Alto, CA: 2007. 1014986.
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10. *Materials Reliability Program: Inspection Standard for PWR Internals – 2012 Update (MRP-228, Rev. 1)*. EPRI, Palo Alto, CA: 2012. 1025147.
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12. Entergy Nuclear Management Manual, EN-DC-202, Rev. 6, "NEI 03-08 Materials Initiative Process," November 21, 2013.
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14. ASME Boiler and Pressure Vessel Code Section XI, 2001 Edition with the 2003 Addenda.
15. U.S. Nuclear Regulatory Commission NUREG-1801, Revision 2, Volumes 1 and 2, "Generic Aging Lessons Learned (GALL) Report," December 2010.
16. U.S. Nuclear Regulatory Commission NUREG-1930, Supplement 1, "Safety Evaluation Report Related to the License Renewal of Indian Point Nuclear Generating Unit Nos. 2 and 3," August 2011.

APPENDIX A
COMPONENT INSPECTION DETAILS

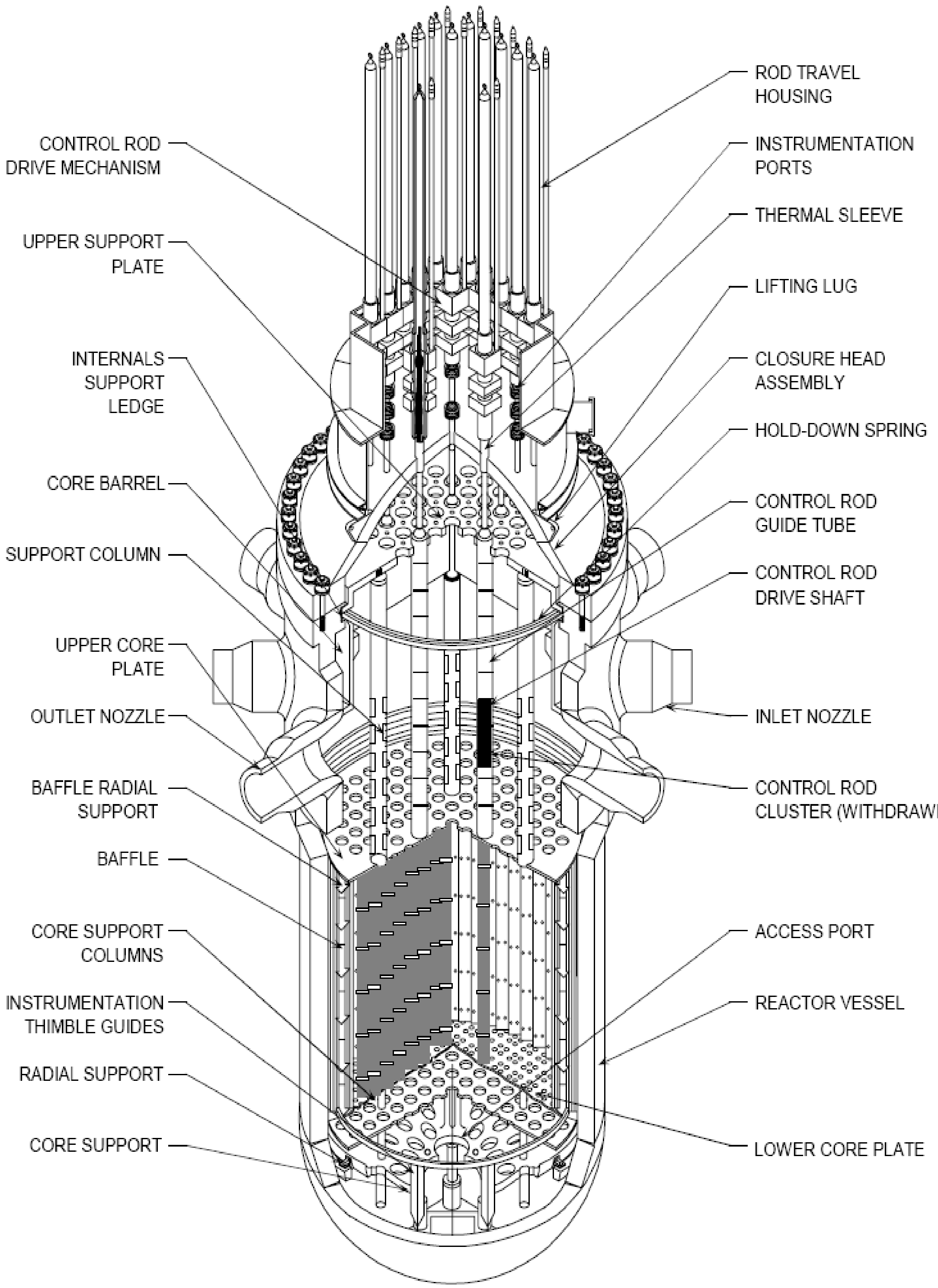


Figure A-1 Typical Westinghouse Internals

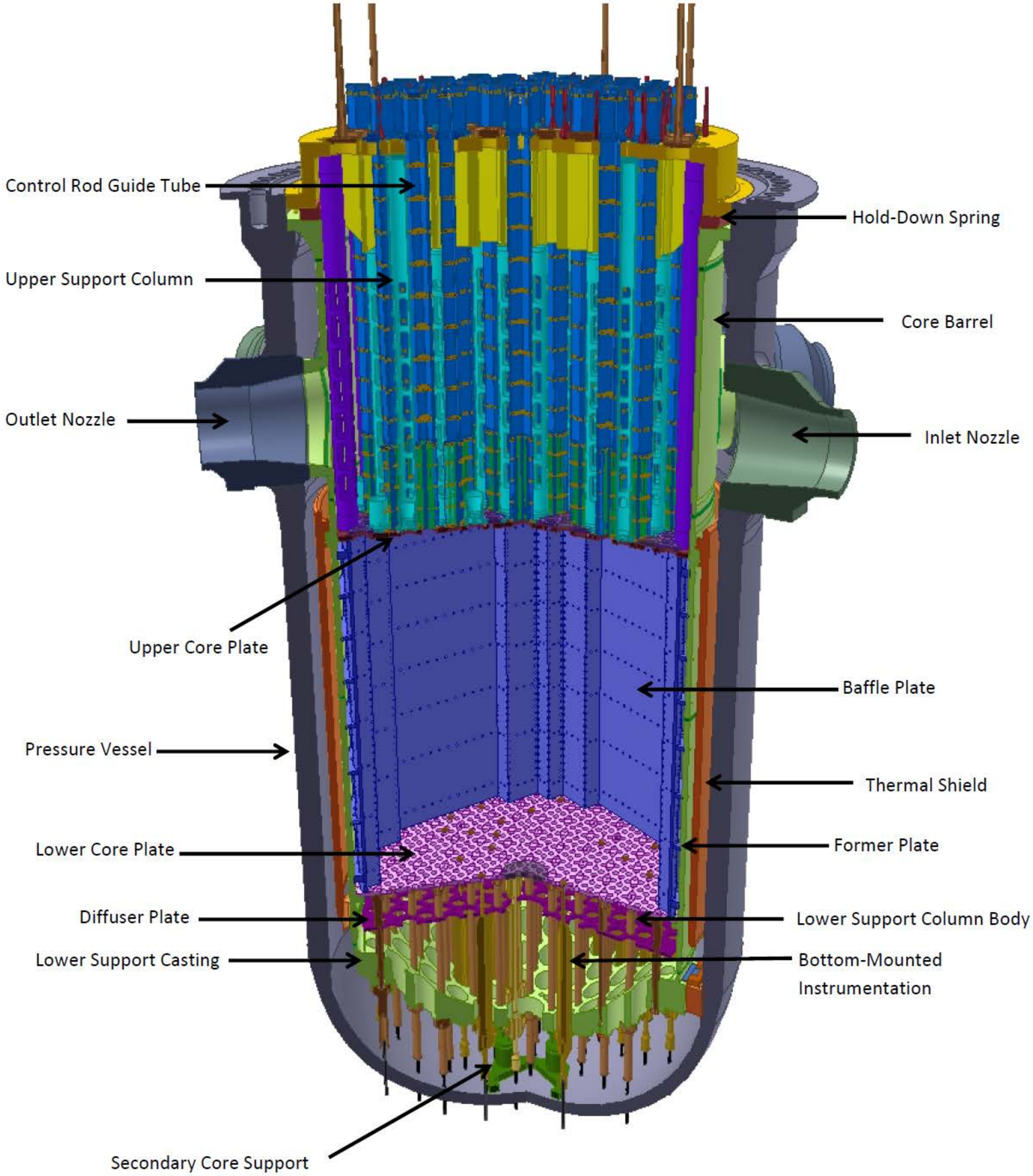
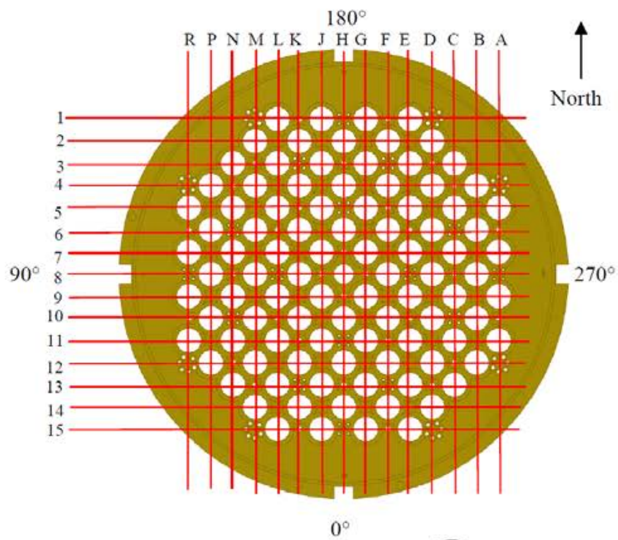
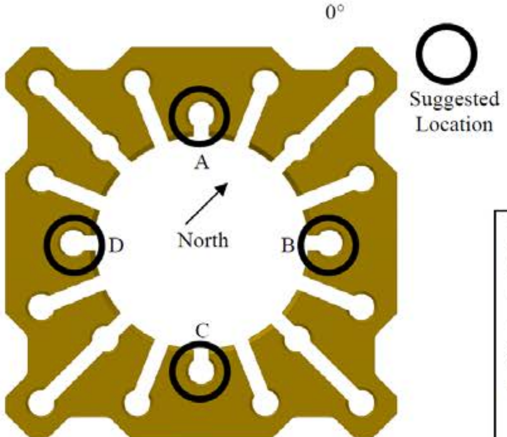


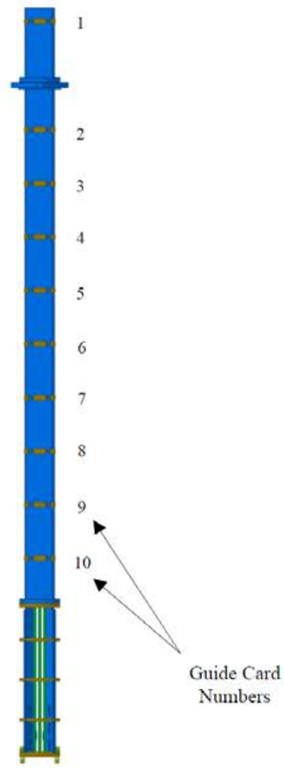
Figure A-2 IP2 Internals

Notes:
1. MRP-227-A/MRP-228 VT-3 Inspection
2. Corresponds to W-ID: 1 in WCAP-17096-NP
3. Base Material: TYPE 304 SS
4. 10 cards per Control Rod Guide Tube; 53 Control Rod Guide Tubes (8 partial guide tubes that do not require inspection at locations: K4, F4, K12, F12, M6, D6, M10, D10). Of the 53 full length guide tubes, only 20% of these (or 11 GTs) require card inspection
5. See WCAP-17894-NP, NL-12-037, IP-RPT-11-00036 for detailed documentation





Guide Card Location Letters (top view)



Naming Convention:
Column Letter - Row Number – Guide Card Number – Location Letter – Ligament Width (inches)

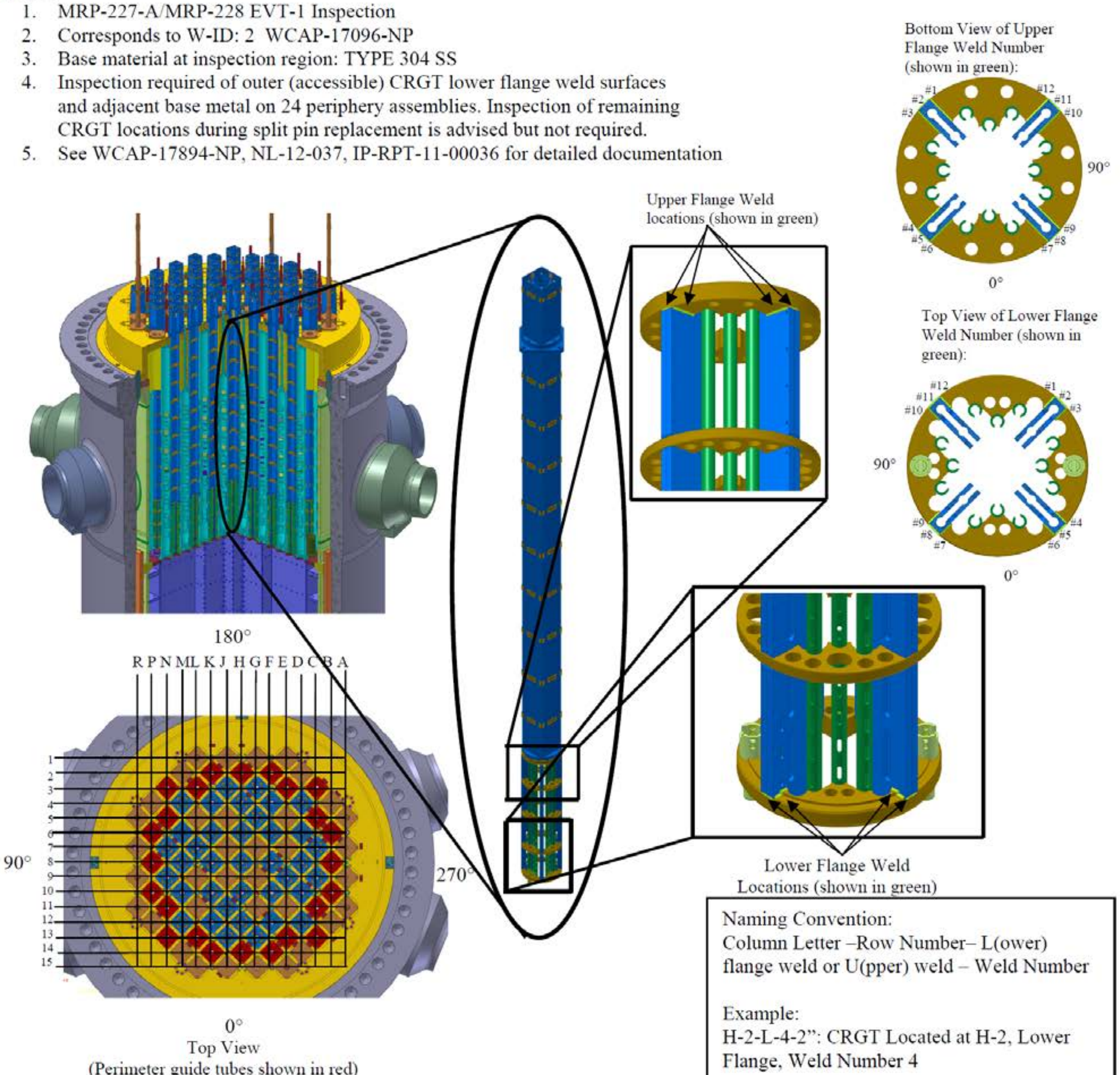
Example:
H-2-3-A-0.277
CRGT location H-2, 3rd guide card from the top, Location A, 0.277 inches

Control Rod Guide Tube Assembly – Guide Plates (Cards)	Westinghouse	
	ENTERGY Indian Point PIEP	
	MRP-227-A COMPONENT INSPECTION DETAIL IMPLEMENTATION SKETCH	
	Parent Document	SKETCH NO.
SCALE: NTS	WCAP-17894-NP, Rev. 0	IP2-CID-0010

Figure A-3 IP2-CID-0010 Control Rod Guide Tube Assembly – Guide Plates (Cards)

Notes:

1. MRP-227-A/MRP-228 EVT-1 Inspection
2. Corresponds to W-ID: 2 WCAP-17096-NP
3. Base material at inspection region: TYPE 304 SS
4. Inspection required of outer (accessible) CRGT lower flange weld surfaces and adjacent base metal on 24 periphery assemblies. Inspection of remaining CRGT locations during split pin replacement is advised but not required.
5. See WCAP-17894-NP, NL-12-037, IP-RPT-11-00036 for detailed documentation



Naming Convention:
Column Letter –Row Number– L(ower) flange weld or U(pper) weld – Weld Number

Example:
H-2-L-4-2": CRGT Located at H-2, Lower Flange, Weld Number 4

Control Rod Guide Tube Assembly – Lower Flange Welds

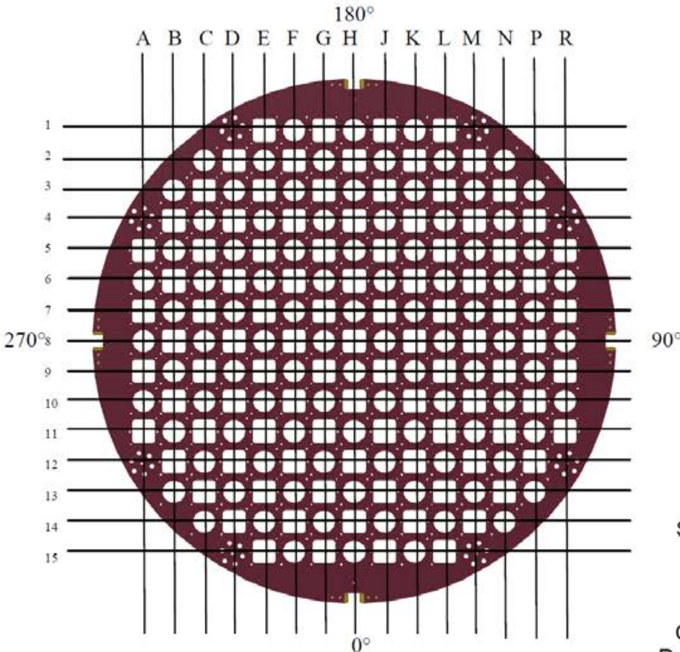
Westinghouse

ENTERGY
Indian Point PIEP
MRP-227-A
COMPONENT INSPECTION DETAIL
IMPLEMENTATION SKETCH

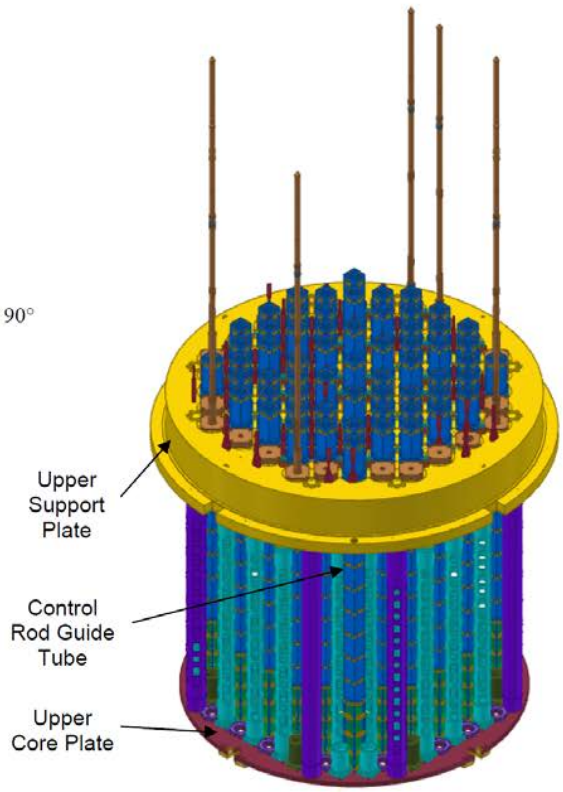
SCALE: NTS	Parent Document	SKETCH NO.
	WCAP-17894-NP, Rev. 0	IP2-CID-0020

Figure A-4 IP2-CID-0020 Control Rod Guide Tube Assembly – Lower Flange Welds

Notes:
1. MRP-227-A/MRP-228 EVT-1 Inspection
2. Corresponds to W-ID: 2.1 in WCAP-17096-NP
3. Material: Type 304 SS
4. 1 Upper Core Plate
5. See WCAP-17894-NP, NL-12-037, IP-RPT-11-00036 for detailed documentation



Upper Core Plate Location Reference
(Bottom View)



Naming Convention:
Column Letter(s) – Row Number(s) - Flaw Length

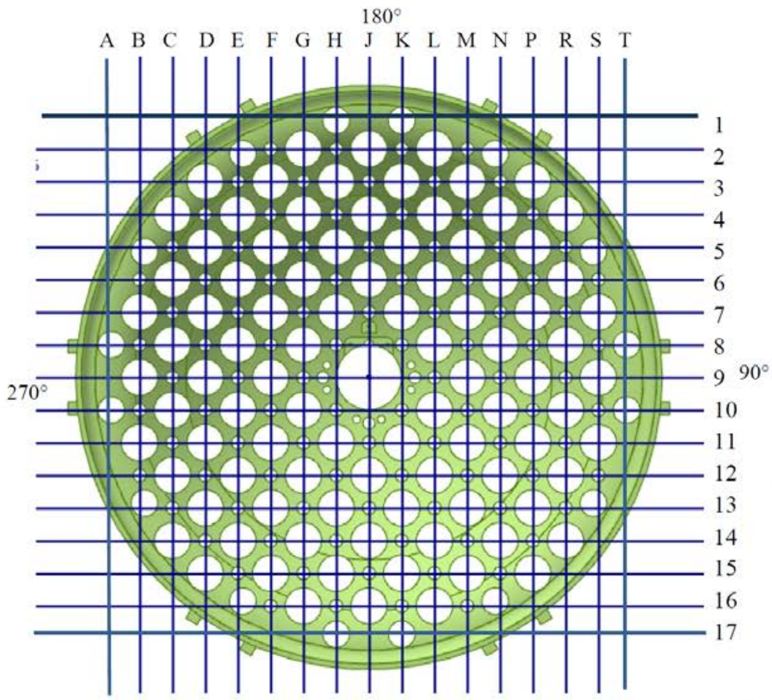
Example One:
M3-1"

Example Two:
Between M3 and M4 – 1"

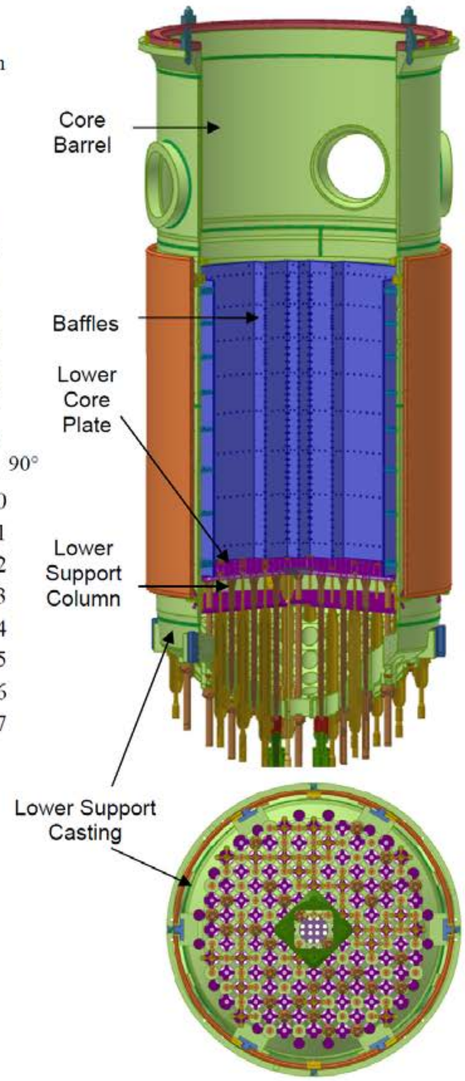
Upper Internals Assembly – Upper Core Plate	Westinghouse	
	ENTERGY Indian Point PIEP	
	MRP-227-A COMPONENT INSPECTION DETAIL IMPLEMENTATION SKETCH	
	Parent Document	SKETCH NO.
SCALE: NTS	WCAP-17894-NP, Rev. 0	IP2-CID-0021

Figure A-5 IP2-CID-0021 Upper Internals Assembly – Upper Core Plate

Notes:
1. MRP-227-A/MRP-228 EVT-1 Inspection
2. Corresponds to W-ID: 2.2 in WCAP-17096-NP
3. Material: Gr. CF8
4. 1 Lower Support Casting
5. See WCAP-17894, NL-12-037, IP-RPT-11-00036 for detailed documentation



Lower Support Casting – Bottom View



Reactor Bottom View (Vessel Not Shown)

Naming Convention:
Column Letter(s) – Row Number(s) - Flaw Length

Example One:
M3-1"

Example Two:
Between M3 and M4 – 1"

Lower Internals Assembly – Lower Support Forging or Castings	Westinghouse	
	ENTERGY Indian Point PIEP	
	MRP-227-A COMPONENT INSPECTION DETAIL IMPLEMENTATION SKETCH	
	Parent Document	SKETCH NO.
SCALE: NTS	WCAP-17894-NP, Rev. 0	IP2-CID-0022

Figure A-6 IP2-CID-0022 Lower Internals Assembly – Lower Support Forging or Castings

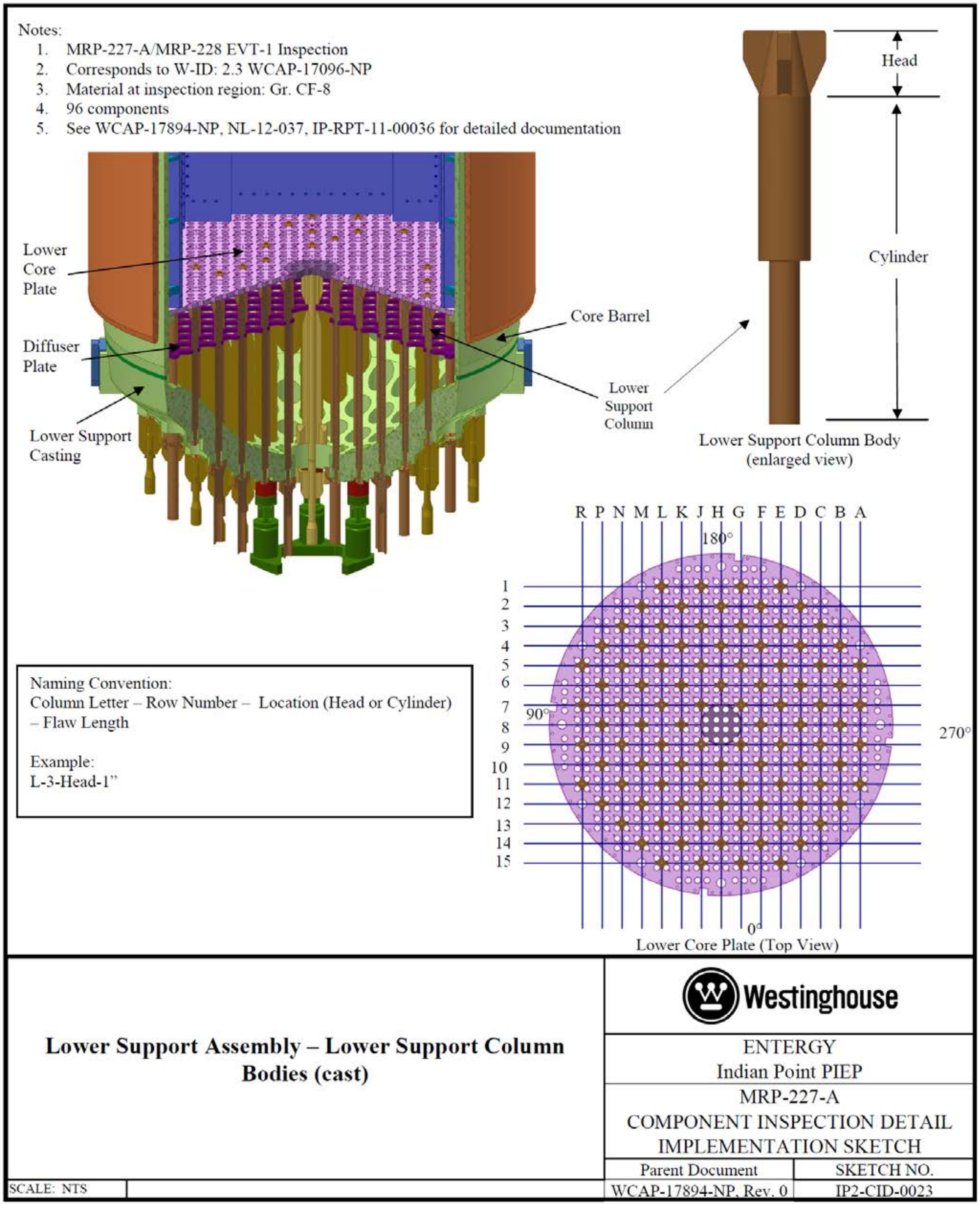
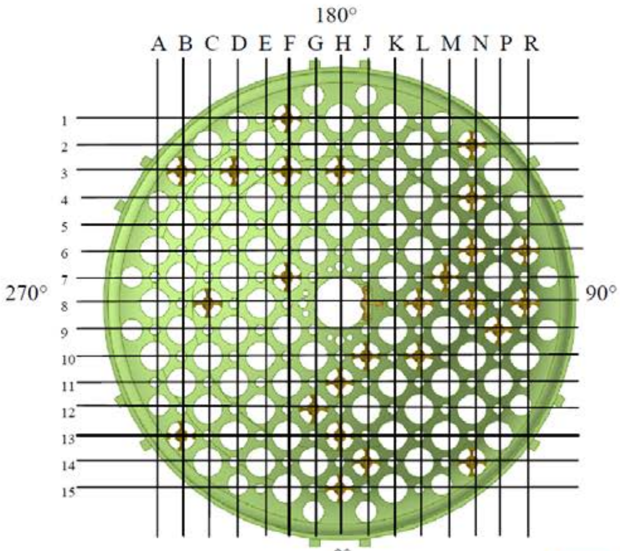


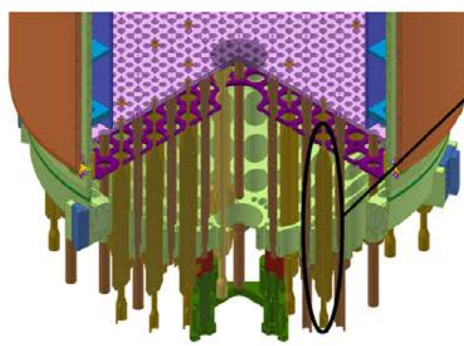
Figure A-7 IP2-CID-0023 Lower Support Assembly – Lower Support Column Bodies (cast)

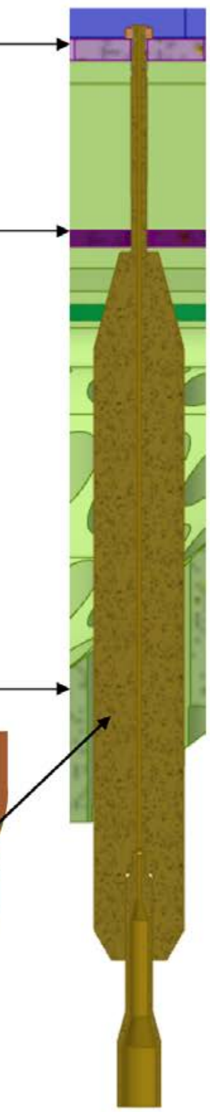
Notes:

1. MRP-227-A/MRP-228 VT-3 Inspection
2. Corresponds to W-ID: 2.4 in WCAP-17096-NP
3. Material: Type 304 SS
4. 26 Bottom Mounted Instrumentation Columns
5. See WCAP-17894-NP, NL-12-037, IP-RPT-11-00036 for detailed documentation



Location Reference
Lower Support Casting
(Bottom View)





Enlarged view – Bottom Mounted Instrumentation Column Body

Naming Convention:
Column Letter – Row Number

Example: M3

Bottom Mounted Instrumentation System – Bottom – Mounted Instrumentation (BMI) Column Bodies

Westinghouse

ENTERGY
Indian Point PIEP

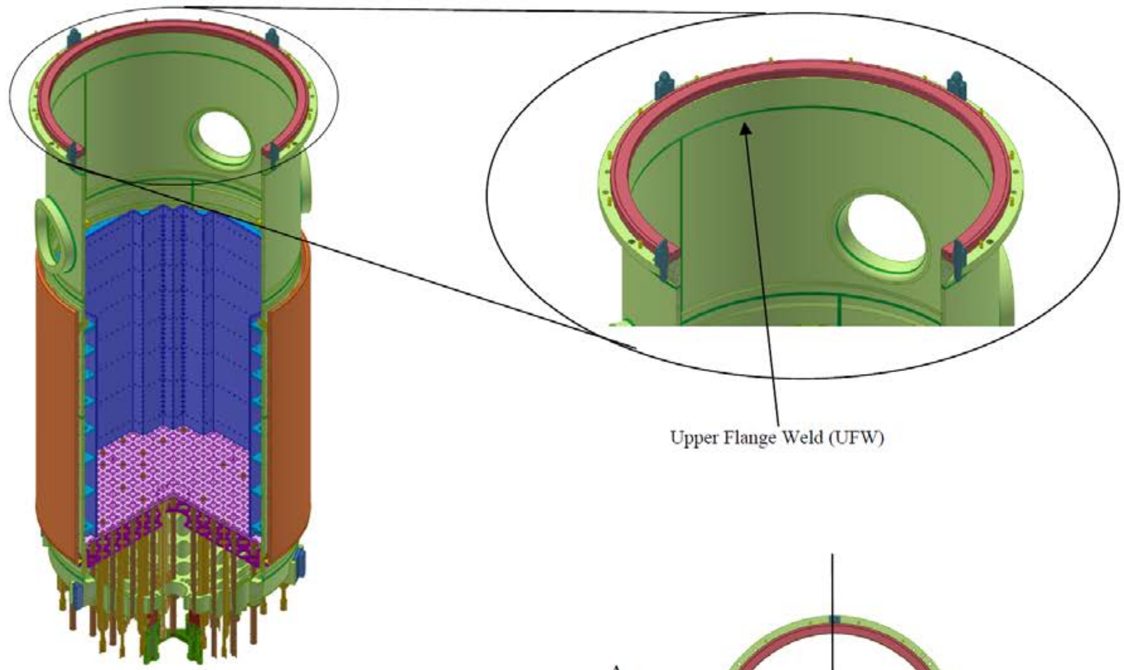
MRP-227-A
COMPONENT INSPECTION DETAIL
IMPLEMENTATION SKETCH

Parent Document	SKETCH NO.
WCAP-17894-NP, Rev. 0	IP2-CID-0024

SCALE: NTS

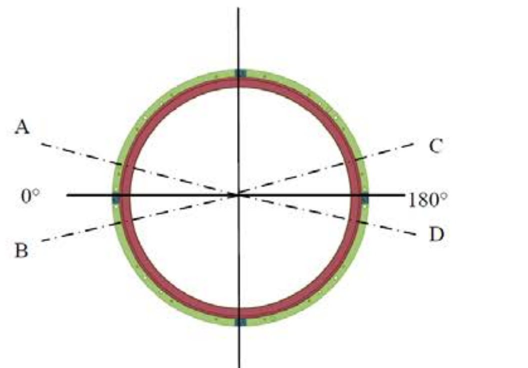
Figure A-8 IP2-CID-0024 Bottom-mounted Instrumentation System – Bottom-mounted Instrumentation (BMI) Column Bodies

Notes:
1. MRP-227-A/MRP-228 EVT-1 Inspection
2. Corresponds to W-ID: 3 WCAP-17096-NP
3. Base material at inspection region: Type 304 SS
4. See WCAP-17894-NP, NL-12-037, IP-RPT-11-00036 for detailed documentation
5. 1 location; may be inspected from inside or outside diameter



Naming convention for observed flaws:
UFW-Length (inches)-Angle at center of flaw (degrees) – flaw on ID or OD

Example:
UFW-10-50- ID would be a 10 inch flaw centered at 50° on the ID

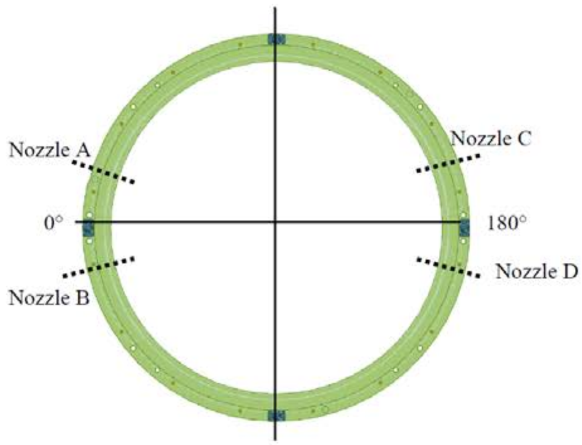


Core Barrel Assembly – Upper Core Barrel Flange Weld	Westinghouse	
	ENTERGY Indian Point PIEP	
	MRP-227-A COMPONENT INSPECTION DETAIL IMPLEMENTATION SKETCH	
	Parent Document	SKETCH NO.
SCALE: NTS	WCAP-17894-NP, Rev. 0	IP2-CID-0030

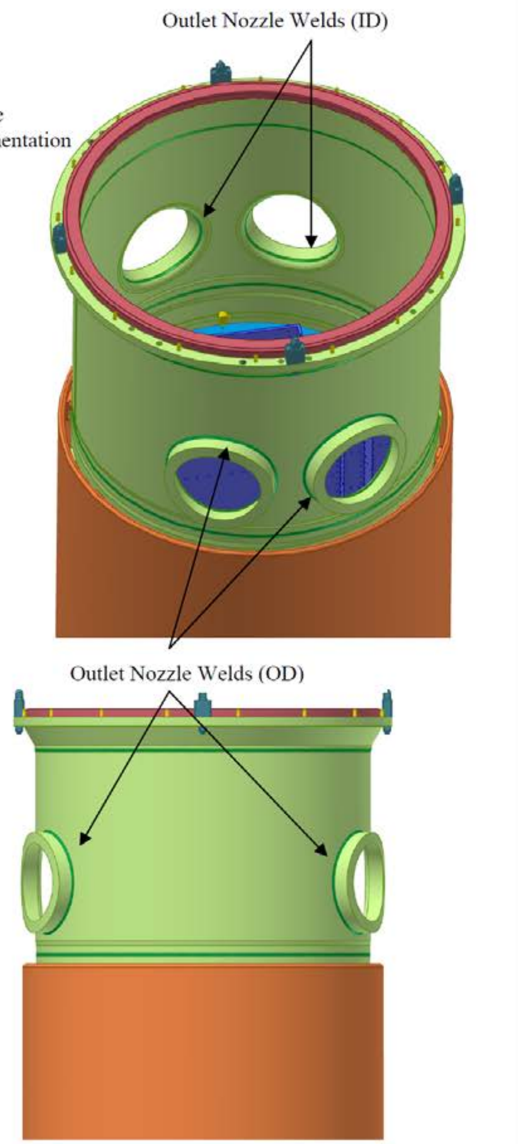
Figure A-9 IP2-CID-0030 Core Barrel Assembly – Upper Core Barrel Flange Weld

Notes:

1. MRP-227-A/MRP-228 EVT-1 Inspection
2. Corresponds to W-ID: 3.1 in WCAP-17096-NP
3. Base material at inspection regions: TYPE 304 SS
4. 4 nozzles, each with an ID and OD weld; Inspect one weld on each nozzle
5. See WCAP-17894-NP, NL-12-037, IP-RPT-11-00036 for detailed documentation



Core Barrel Top View –
Approximate Nozzle Locations



Naming Convention:
Weld Name – Length of Flaw (inches) – Angle at center of flaw - ID or OD

Example:
Nozzle A-10-30-OD would indicate a 10 inch flaw centered at 30° on the OD of Nozzle Weld A.

Core Barrel Assembly – Core Barrel Outlet Nozzle Welds

Westinghouse

ENTERGY
Indian Point PIEP

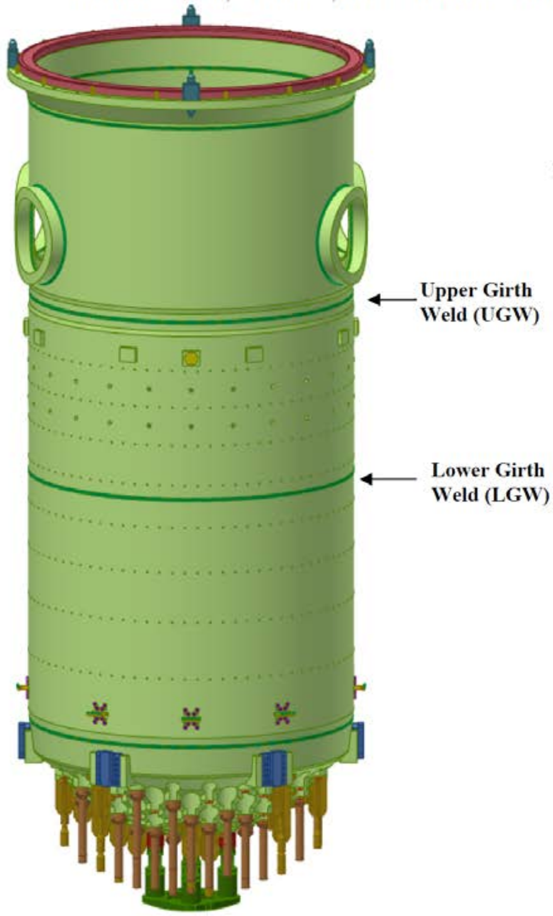
MRP-227-A
COMPONENT INSPECTION DETAIL
IMPLEMENTATION SKETCH

Parent Document	SKETCH NO.
WCAP-17894-NP, Rev. 0	IP2-CID-0031

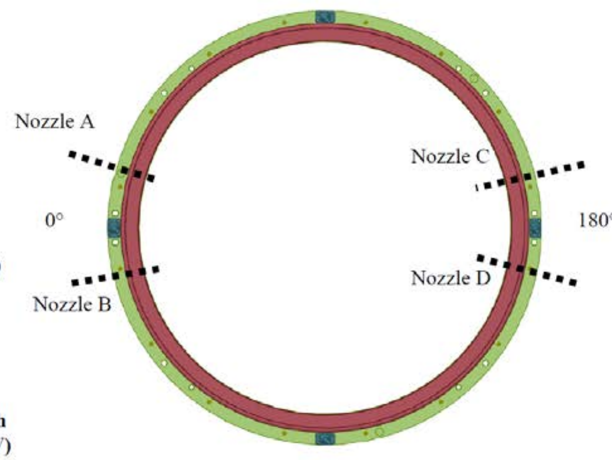
SCALE: NTS

Figure A-10 IP2-CID-0031 Core Barrel Assembly – Core Barrel Outlet Nozzle Welds

Notes:
1. MRP-227-A/MRP-228 EVT-1 Inspection
2. Corresponds to W-ID: 4 in WCAP-17096-NP
3. Base material at inspection region: TYPE 304 SS
4. 2 locations, may be inspected from ID or OD of core barrel
5. See WCAP-17894-NP, NL-12-037, IP-RPT-11-00036 for detailed documentation



Core Barrel Assembly
(Thermal shield removed for clarity)



Core Barrel Top View –
Approximate Nozzle Locations

Naming Convention:
Weld Name – Length of Flaw (inches) – Angle at center of flaw - ID or OD

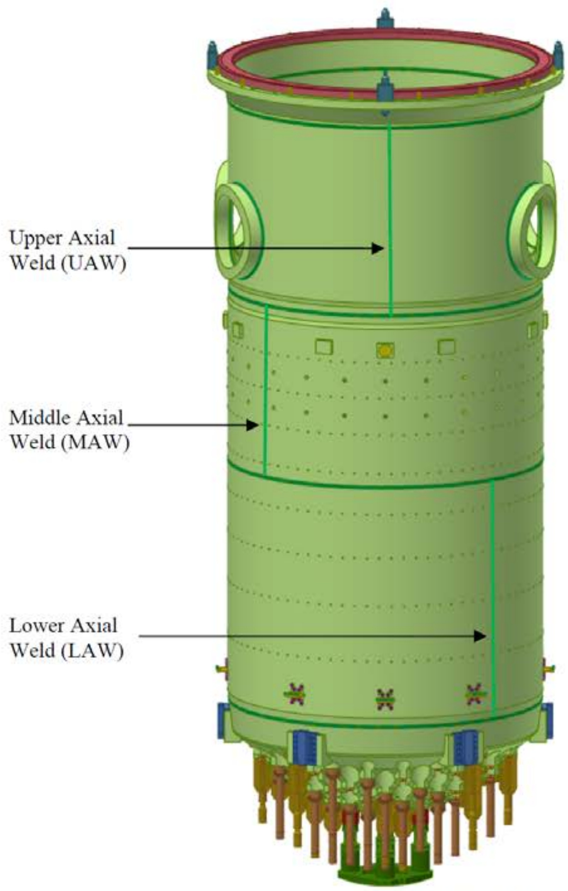
Example:
LGW-10-30-OD
10 inch flaw centered at 30° on the OD of the lower girth weld

Core Barrel Assembly – Upper and Lower Core Barrel Cylinder Girth Welds	Westinghouse	
	ENTERGY Indian Point PIEP	
	MRP-227-A COMPONENT INSPECTION DETAIL IMPLEMENTATION SKETCH	
	Parent Document	SKETCH NO.
SCALE: NTS	WCAP-17894-NP, Rev. 0	IP2-CID-0040

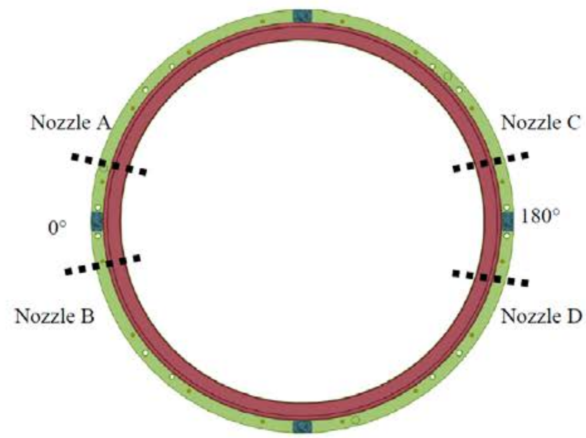
Figure A-11 IP2-CID-0040 Core Barrel Assembly – Upper and Lower Core Barrel Cylinder Girth Welds

Notes:

1. MRP-227-A/MRP-228 EVT-1 Inspection
2. Corresponds to W-ID: 4.1 in WCAP-17096-NP
3. Base material at inspection region: TYPE 304 SS
4. 3 locations, may be inspected from ID or OD of core barrel
5. See WCAP-17894, NL-12-037, IP-RPT-11-00036 for detailed documentation



Core Barrel Assembly – Axial Welds Rotated into View
(Thermal Shield Removed)



Core Barrel Top View –
Approximate Nozzle Locations

Naming Convention:
Weld Name – Length of Flaw (inches) – Distance from top of Weld to center of flaw - ID or OD

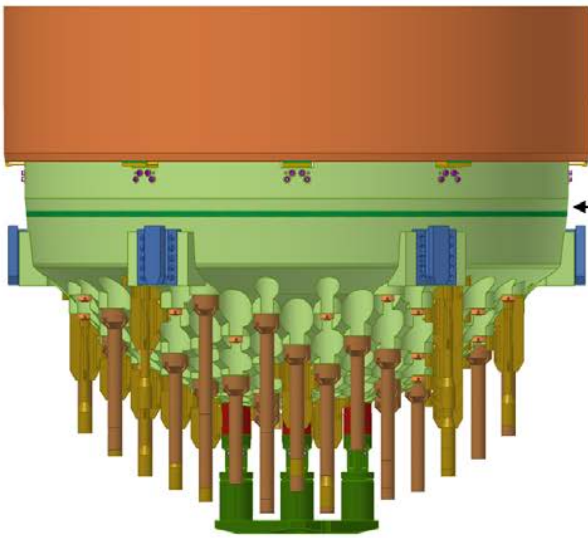
Example:
LAW-10-30-OD
10 inch flaw centered at 30'' on the OD of the lower axial weld

Core Barrel Assembly – Upper and Lower Core Barrel Cylinder Axial Welds	Westinghouse	
	ENTERGY Indian Point PIEP	
	MRP-227-A COMPONENT INSPECTION DETAIL IMPLEMENTATION SKETCH	
	Parent Document	SKETCH NO.
SCALE: NTS	WCAP-17894-NP, Rev. 0	IP2-CID-0041

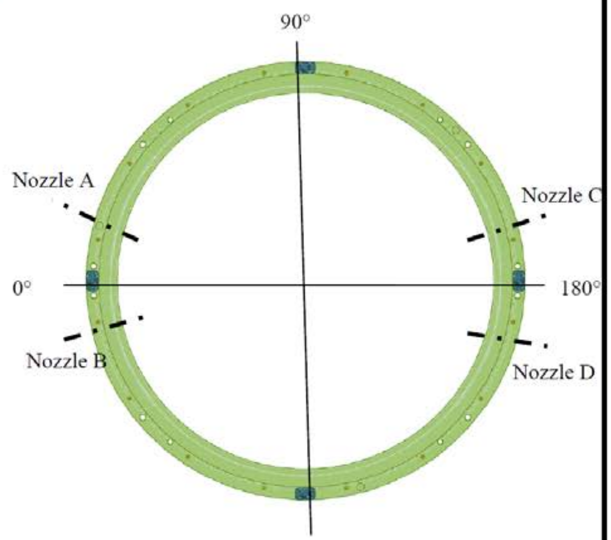
Figure A-12 IP2-CID-0041 Core Barrel Assembly – Upper and Lower Core Barrel Cylinder Axial Welds

Notes:

1. MRP-227-A/MRP-228 EVT-1 Inspection
2. Corresponds to W-ID: 5 in WCAP-17096-NP
3. Base material at inspection regions: TYPE 304 SS (Core Barrel) and Gr. CF8 (Lower Core Support Casting)
4. 1 location; May inspect from ID or OD
5. See WCAP-17894-NP, NL-12-037, IP-RPT-11-00036 for detailed documentation



Lower Flange Weld (LFW)
– Shown in Green



Core Barrel Top View –
Approximate Nozzle Locations

Naming Convention:
Weld Name – Length of Flaw (inches) – Angle at center of flaw - ID or OD

Example:
LFW-10-30-OD
10 inch flaw centered at 30° on the OD of the lower flange weld

Core Barrel Assembly - Lower Core Barrel Flange Weld	Westinghouse	
	ENTERGY Indian Point PIEP	
	MRP-227-A COMPONENT INSPECTION DETAIL IMPLEMENTATION SKETCH	
	Parent Document	SKETCH NO.
SCALE: NTS	WCAP-17894-NP, Rev. 0	IP2-CID-0050

Figure A-13 IP2-CID-0050 Core Barrel Assembly – Lower Core Barrel Flange Weld

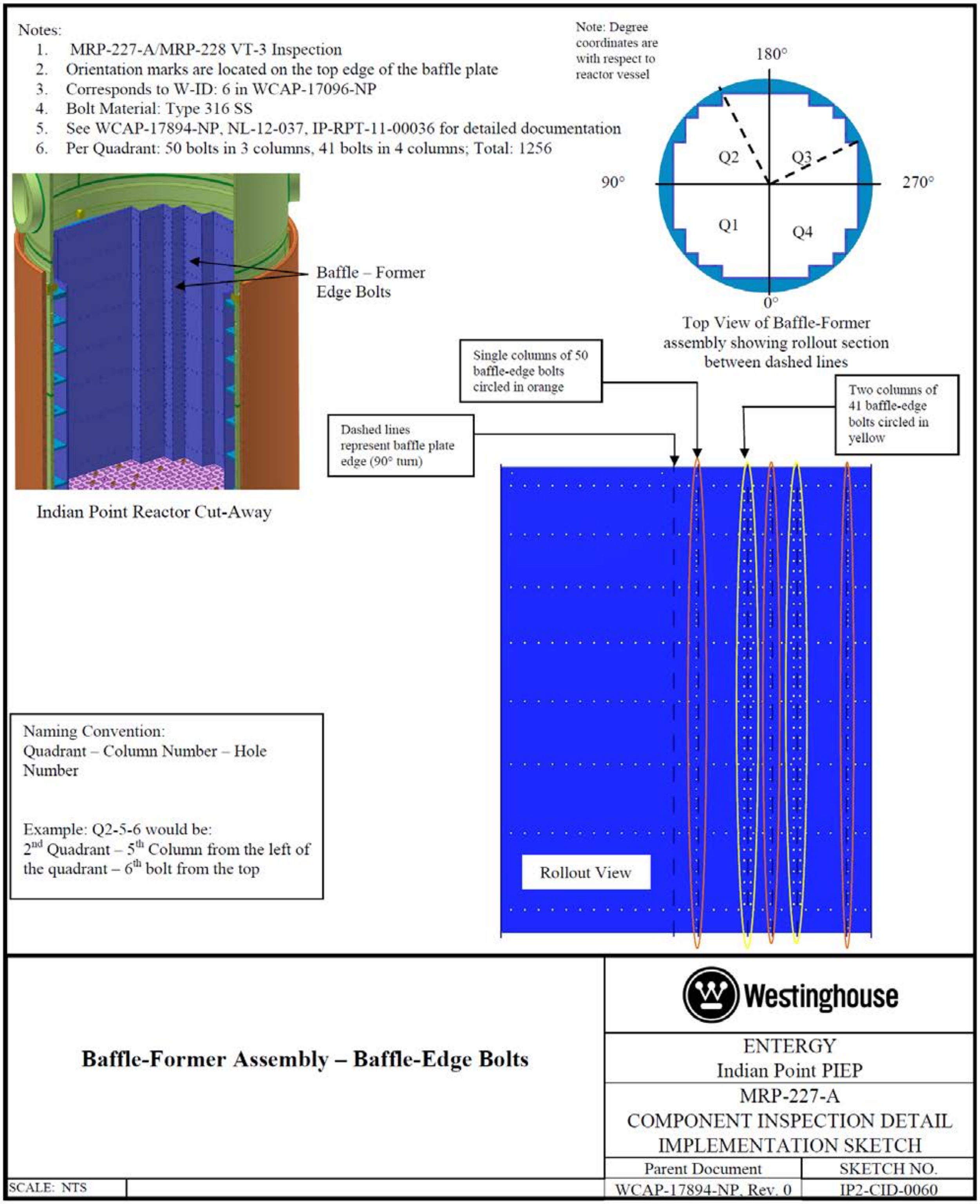


Figure A-14 IP2-CID-0060 Baffle-former Assembly – Baffle-edge Bolts

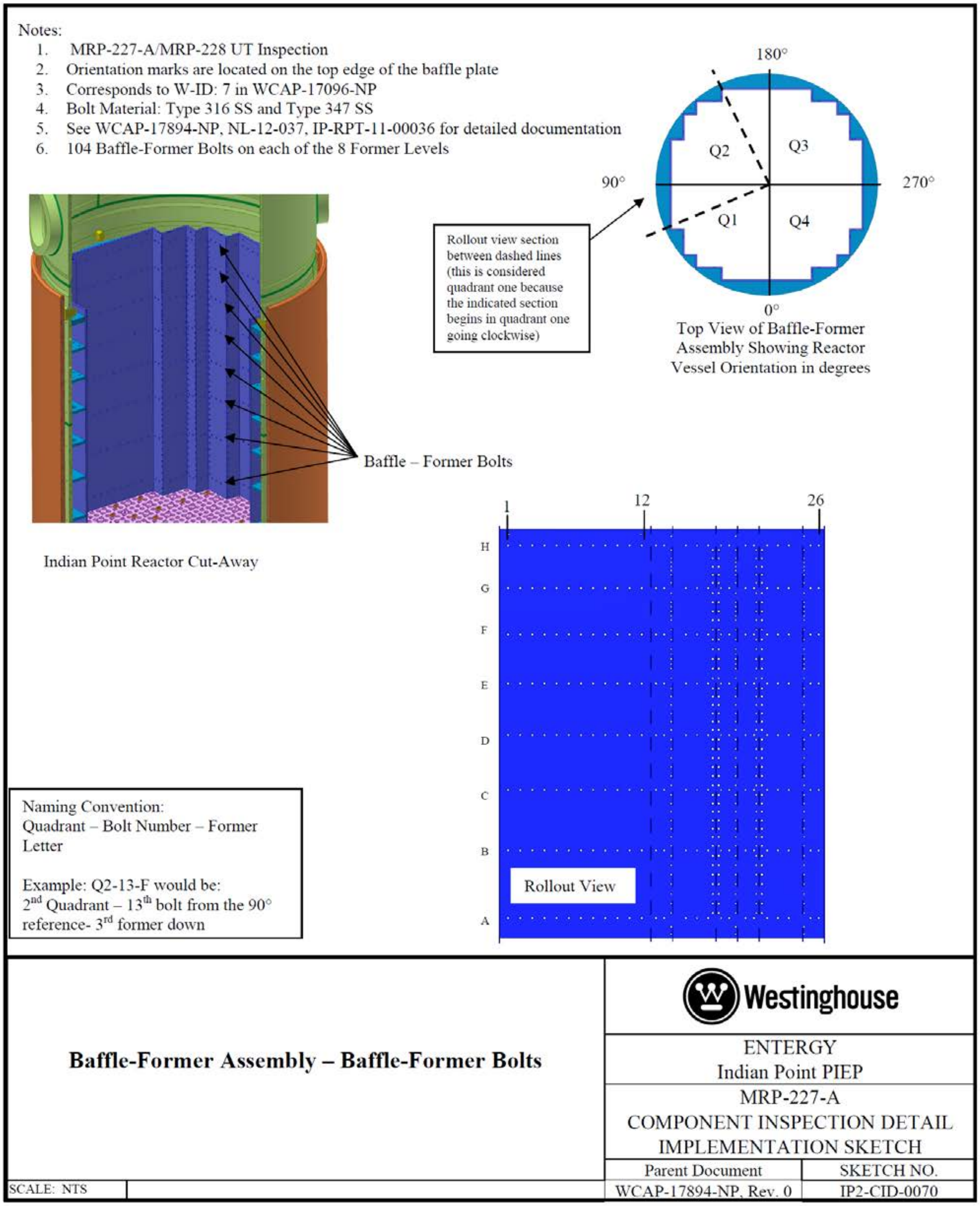


Figure A-15 IP2-CID-0070 Baffle-former Assembly – Baffle-former Bolts

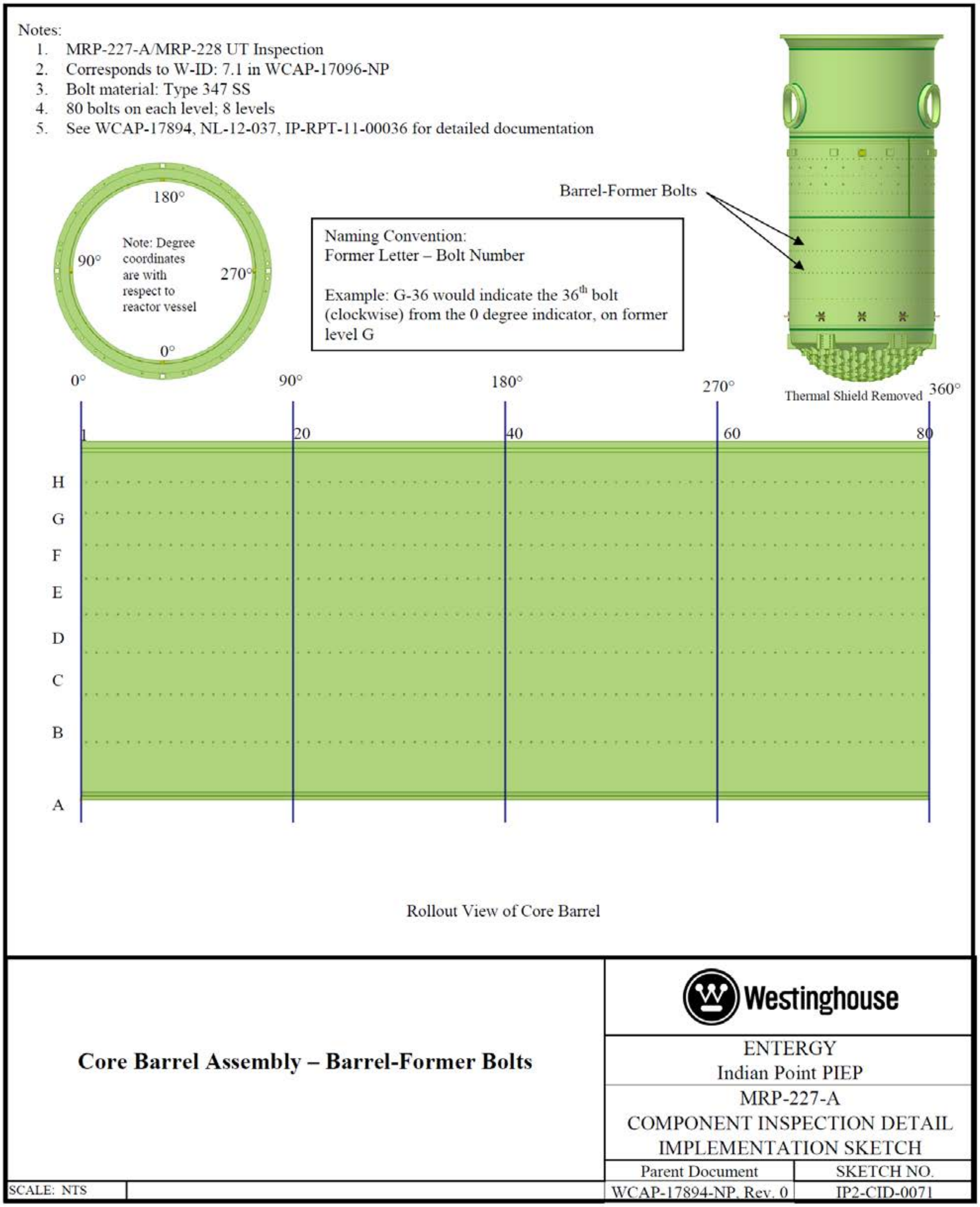
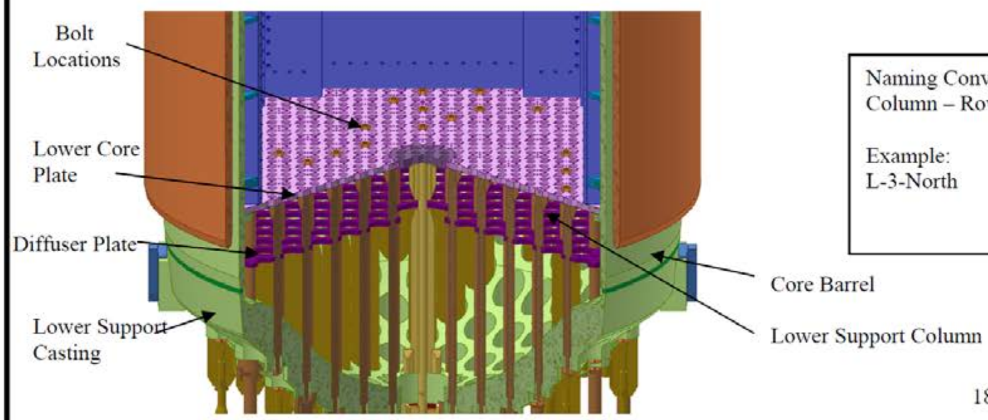


Figure A-16 IP2-CID-0071 Core Barrel Assembly – Barrel-former Bolts

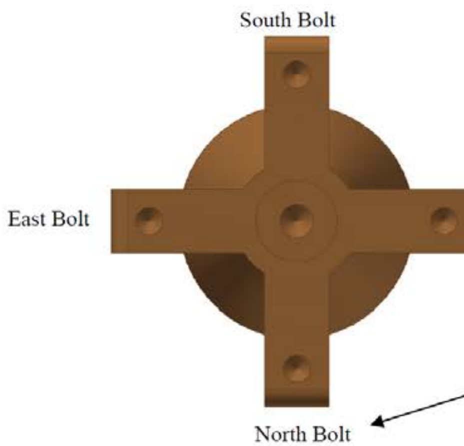
Notes:

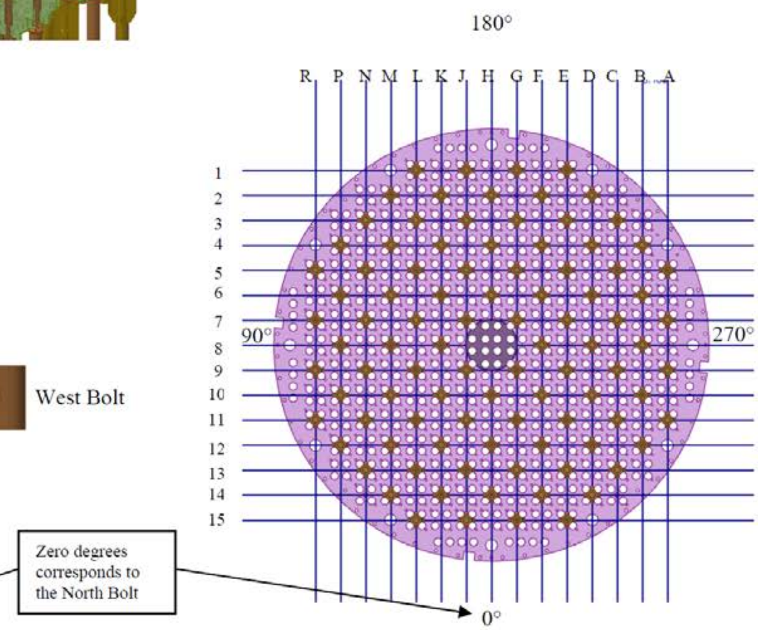
1. MRP-227-A/MRP-228 UT Inspection
2. Corresponds to W-ID: 7.2 in WCAP-17096-NP
3. Bolt Material: Type 316 SS
4. 384 bolts
5. See WCAP-17894-NP, NL-12-037, IP-RPT-11-00036 for detailed documentation



Naming Convention:
Column – Row – Bolt Location

Example:
L-3-North





Lower Core Plate (Top View) – support column locations shown

Lower Support Assembly – Lower Support Column Bolts	Westinghouse	
	ENTERGY Indian Point PIEP	
	MRP-227-A COMPONENT INSPECTION DETAIL IMPLEMENTATION SKETCH	
	Parent Document	SKETCH NO.
SCALE: NTS	WCAP-17894-NP, Rev. 0	IP2-CID-0072

Figure A-17 IP2-CID-0072 Lower Support Assembly – Lower Support Column Bolts

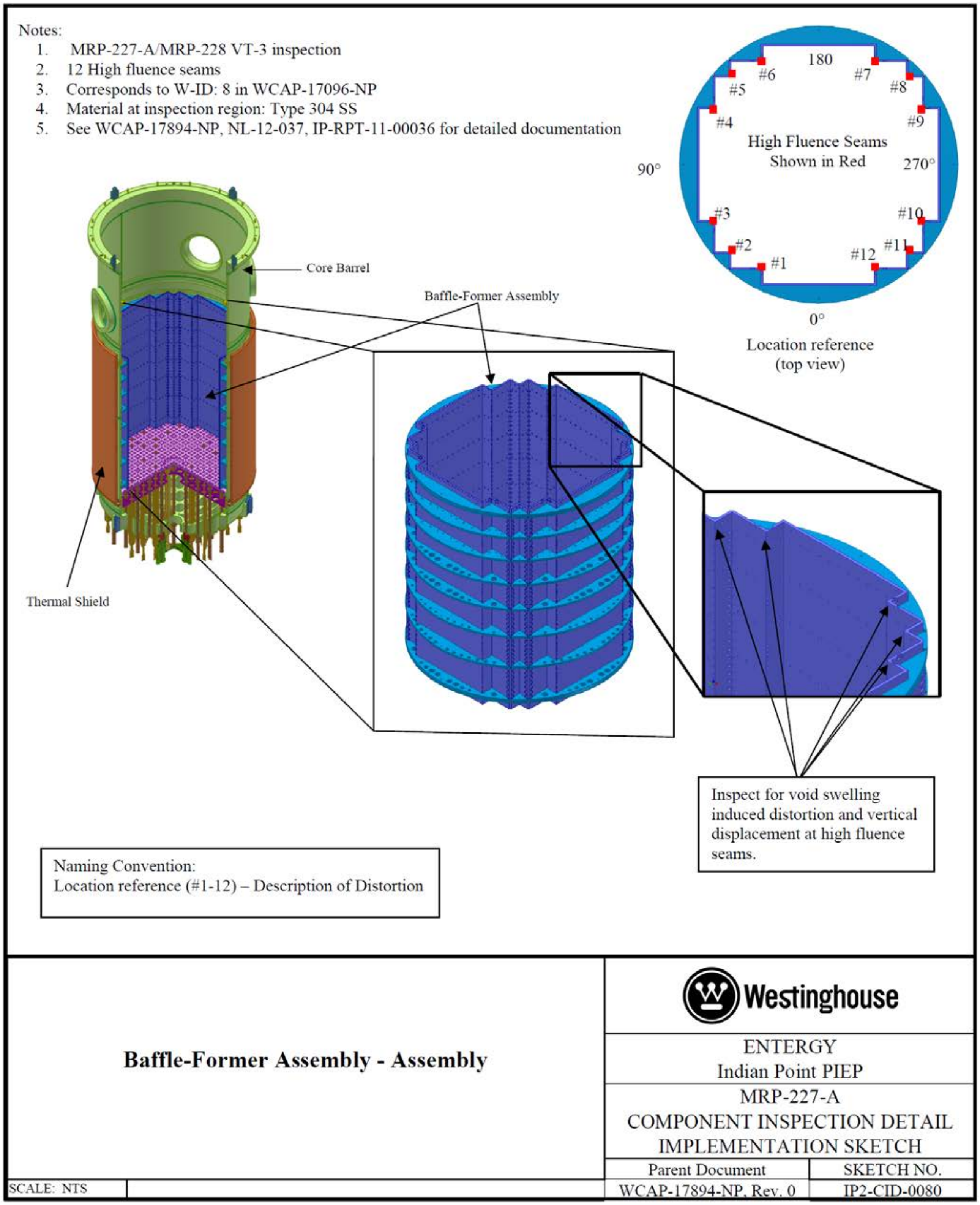
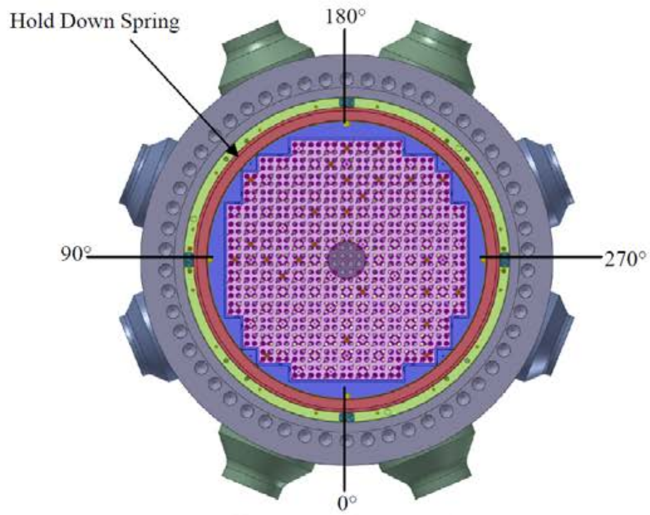
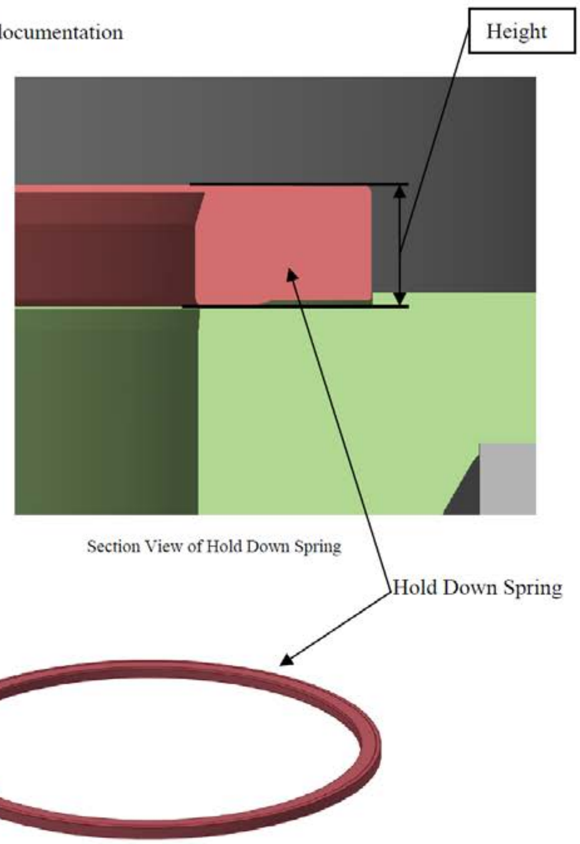


Figure A-18 IP2-CID-0080 Baffle-former Assembly – Assembly

Notes:
1. MRP-227-A/MRP-228 Direct Measurement Inspection
2. 1 Spring, Recommend 8 locations (equally spaced) for measurements, 3 measurements at each location
3. Measure height of hold down spring
4. Corresponds to W-ID: 9 in WCAP-17096-NP
5. Material at inspection region: Grade 304 SS
6. See WCAP-17894-NP, NL-12-037, IP-RPT-11-00036 for detailed documentation



Hold Down Spring Orientation
(Top View)



Section View of Hold Down Spring

Naming Convention:
Angular Orientation Measuring Clockwise from 0 (°) – Height of Spring (inches)+- Measurement Uncertainty

Example:
32-1.25 +- 0.01 would be a height of 1.25+-0.01 inches measured at 32°

Alignment and Interfacing Components – Internals Hold Down Spring	Westinghouse
	ENTERGY Indian Point PIEP
	MRP-227-A COMPONENT INSPECTION DETAIL IMPLEMENTATION SKETCH
SCALE: NTS	Parent Document: WCAP-17894-NP, Rev. 0 — SKETCH NO.: IP2-CID-0090

Figure A-19 IP2-CID-0090 Alignment and Interfacing Components – Internals Hold-down Spring

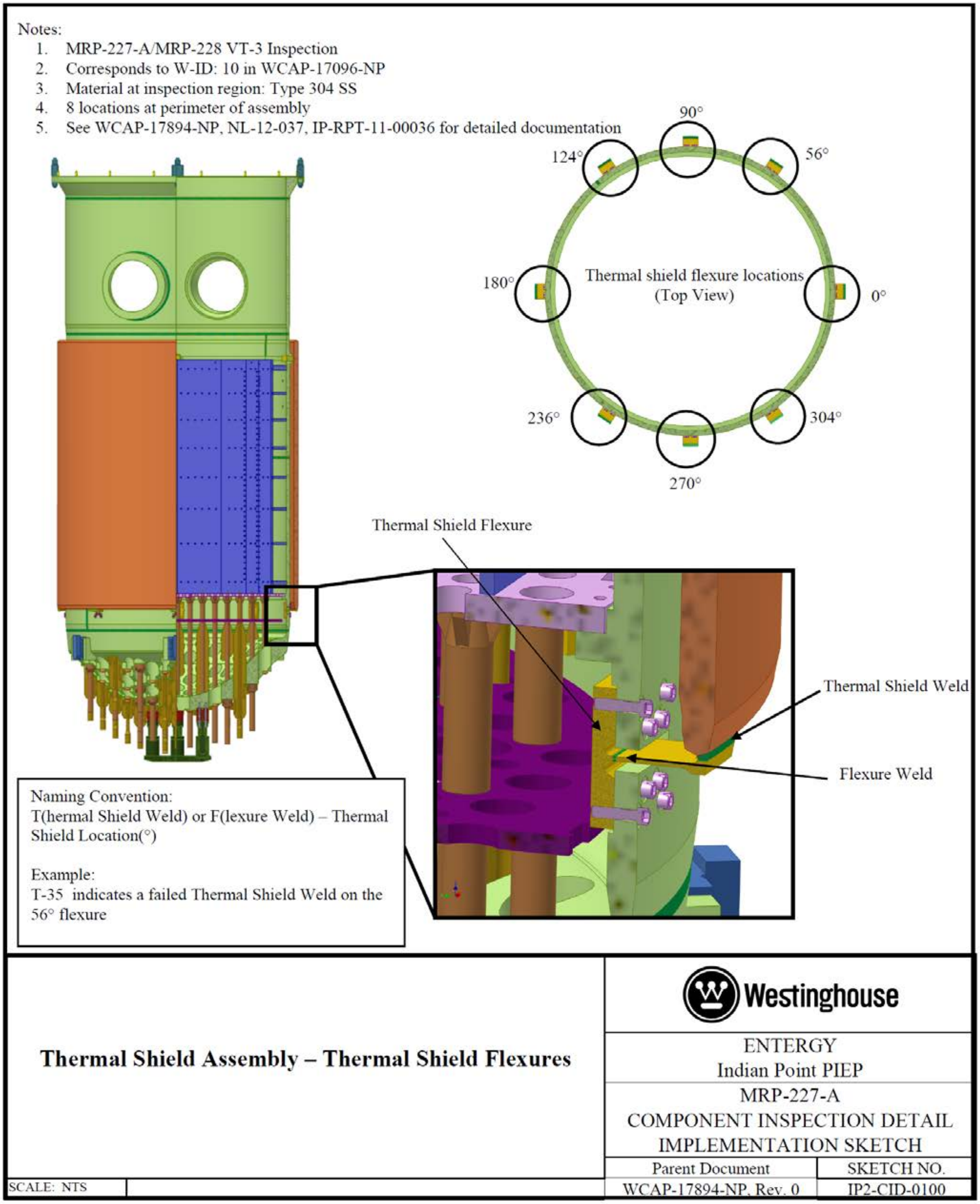


Figure A-20 IP2-CID-0100 Thermal Shield Assembly – Thermal Shield Flexures