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NL-17-020

February 6, 2017

U.S. Nuclear Regulatory Commission  
Document Control Desk  
11545 Rockville Pike, TWFN-2 F1  
Rockville, MD 20852-2738

**SUBJECT:** License Renewal Application – Revisions to Reactor Vessel Internals  
Aging Management Program and Inspection Plan  
Indian Point Nuclear Generating Unit Nos. 2 and 3  
Docket Nos. 50-247 and 50-286 (License Nos. DPR-26 and DPR-64)

**REFERENCES:**

- 1) Entergy Letter NL-07-039, "Indian Point Energy Center License Renewal Application" (Apr. 23, 2007) (ML071210507)
- 2) Entergy Letter NL-10-063, "Amendment 9 to License Renewal Application (LRA) – Reactor Vessel Internals Program" (July 14, 2010) (ML102010102)
- 3) Entergy Letter NL-11-107, "License Renewal Application – Completion of Commitment # 30 Regarding the Reactor Vessel Internals Inspection Plan" (Sept. 28, 2011) (ML11280A121)
- 4) Electric Power Research Institute, MRP-227-A, "Materials Reliability Program: Pressurized Water Reactor Internals Inspection and Evaluation Guidelines" (Dec. 2011) (ML120170453)
- 5) Entergy Letter NL-12-037, "License Renewal Application – Revised Reactor Vessel Internals Program and Inspection Plan Compliant with MRP-227-A" (Feb. 17, 2012) (ML12060A312)
- 6) NUREG-1930, Supp. 2, "Safety Evaluation Report Related to the License Renewal of Indian Point Nuclear Generating Unit Nos. 2 and 3" (Nov. 2014) (ML15188A383)
- 7) Entergy Letter NL-16-053, "License Event Report # 2016-004-00, Unanalyzed Condition Due to Degraded Reactor Baffle-Former Bolts" (May 31, 2016) (ML16159A219)

Dear Sir or Madam:

By letter dated April 23, 2007 (Reference 1), Entergy Nuclear Operations, Inc. (Entergy) submitted an application pursuant to 10 CFR Part 54 and 10 CFR Part 51, to renew the operating licenses for Indian Point Nuclear Generating Unit Nos. 2 and 3 (IP2 and IP3), for review by the U.S. Nuclear Regulatory Commission (NRC). Entergy provided a description of the Indian Point Energy Center (IPEC) Reactor Vessel Internals (RVI) aging management program (AMP) in Amendment 9 to the License Renewal Application (LRA) (Reference 2).

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Consistent with License Renewal Commitment 30, Entergy submitted its Reactor Vessel Internals (RVI) Inspection Plan on September 28, 2011, two years prior to entering the period of extended operation (PEO) for IP2 (Reference 3). Although only IP2 was within two years of entering the PEO at that time, the RVI Inspection Plan covered both units.

Entergy submitted the RVI Inspection Plan based on the new aging management program (AMP) in NUREG-1801, Revision 2. After the Electric Power Research Institute (EPRI) issued the NRC-approved generic industry aging management guidance for RVIs in MRP-227-A (Reference 4), Entergy submitted a revised RVI AMP and Inspection Plan for both IP2 and IP3 based on MRP-227-A on February 17, 2012 (Reference 5). Following Entergy's submission of additional technical information in response to Staff requests for additional information, the Staff approved Entergy's revised RVI AMP and Inspection Plan, as documented in Safety Evaluation Report Supplement 2 issued in November 2014 (Reference 6).

During the Spring 2016 IP2 refueling outage (2R22), Entergy performed ultrasonic (UT) examinations and/or visual inspections of all 832 baffle-former bolts (bolts) in accordance with the MRP-227-A guidelines. As a result of the inspection findings, Entergy replaced all 227 bolts with actual and assumed indications. It also replaced an additional 51 bolts to reduce the probability of future failures as well as minimize the probability of clusters of failed bolts, resulting in a total of 278 replaced bolts. See Reference 7.

As a result of the IP2 inspection findings and other industry operating experience (OE) indicating a significant number of failed bolts at other similarly-designed PWR plants, Entergy recently revised portions of the Indian Point Energy Center (IPEC) PWR Vessel Internals Program (SEP-PVI-IPEC-001, Rev. 1). The revisions included the addition of new Section 6.2 to incorporate discussion of the recent Unit 2 OE described above, including Entergy's related decision to arrange for offsite fractographic examination of eight baffle-former bolts removed from the IP2 baffle structure during the Spring 2016 outage. The revisions also reflect changes to Entergy's schedule and plans for conducting future UT and visual inspections as well as replacement of baffle-former bolts at IP2 and IP3.

The purpose of this submittal is to make corresponding revisions to the IPEC RVI AMP and IPEC Reactor Vessel Internals Plan, as submitted to the NRC on February 17, 2012. The revisions include: (1) an updated discussion of IPEC and industry operating experience involving baffle-former bolts in Section B.1.42 (Reactor Vessel Internals Program) of the LRA; (2) the addition of new Section 6.0 (Operation Experience and Additional Considerations), which coincides with current Section 6.0 of SEP-PVI-IPEC-001, to the IPEC Reactor Vessel Internals Plan; and (3) a related revision to Table 5-2 (Primary Components at IPEC Units 2 and 3) of the IPEC Reactor Vessel Internals Plan to cross-reference newly-added Section 6.2 (Spring 2016 Operating Experience) and the IPEC-specific baffle-former bolt examination/inspection plans discussed therein.

There are no new commitments identified in this submittal. If you have any questions, or require additional information, please contact Mr. Robert Walpole at 914-254-6710.

I declare under penalty of perjury that the foregoing is true and correct. Executed on  
2/6, 2017.

Sincerely,



AJV/rl

- Attachments:
1. Indian Point Energy Center Revised Reactor Vessel Internals Program.
  2. Indian Point Energy Center Revised Reactor Vessel Internals Inspection Plan

cc: Mr. Daniel H. Dorman, Regional Administrator, NRC Region I  
Mr. Sherwin E. Turk, NRC Office of General Counsel, Special Counsel  
Mr. William Burton, NRC Senior Project Manager, Division of License Renewal  
Mr. Douglas Pickett, NRR Senior Project Manager  
Ms. Bridget Frymire, New York State Department of Public Service  
Mr. John B. Rhodes, President and CEO NYSERDA  
NRC Resident Inspector's Office

**ATTACHMENT 1 TO NL-17-020**

**INDIAN POINT ENERGY CENTER**

**REVISED REACTOR VESSEL INTERNALS PROGRAM**

Additions Underlined  
Deletions Lined Out

**ENERGY NUCLEAR OPERATIONS, INC.  
INDIAN POINT NUCLEAR GENERATING UNIT NOS. 2 AND 3  
DOCKET NOS. 50-247 AND 50-286**

#### **A.2.1.41 Reactor Vessel Internals Aging Management Activities**

The Reactor Vessel Internals (RVI) Program is a new plant specific program to manage aging effects of reactor vessel internals using the guidance from the Electric Power Research Institute (EPRI) Materials Reliability Program (MRP). The MRP inspection and evaluation (I&E) guidelines for managing the effects of aging on pressurized water reactor vessel internals are presented in MRP-227-A, "Materials Reliability Program: Pressurized Water Reactor Internals Inspection and Evaluation Guidelines." The MRP also developed inspection requirements specific to the inspection methods delineated in MRP-227-A, as well as requirements for qualification of the nondestructive examination (NDE) systems used to perform those inspections. These inspection requirements are presented in MRP-228, "Materials Reliability Program: Inspection Standard for PWR Internals."

MRP-227-A and MRP-228 provide the basis of the IPEC Reactor Vessel Internals (RVI) Program. The RVI Program will monitor the effects of aging degradation mechanisms on the intended functions of the internals through periodic and conditional examinations. The RVI Program will detect and evaluate cracking, loss of material, reduction of fracture toughness, loss of preload and dimensional changes of vessel internals components in accordance with MRP-227-A inspection requirements and evaluation acceptance criteria.

The IPEC RVI Program will be implemented and maintained in accordance with the guidance in NEI 03-08 [Addenda], Addendum A, "RCS Materials Degradation Management Program Guidelines." Any deviations from mandatory, needed, or good practice implementation requirements established in MRP-227-A or MRP-228, will be resolved in accordance with the NEI 03-08 implementation protocol. The RVI Program will be implemented prior to the period of extended operation.

#### **A.3.1.41 Reactor Vessel Internals Aging Management Activities**

The Reactor Vessel Internals (RVI) Program is a new plant specific program to manage aging effects of reactor vessel internals using the guidance from the Electric Power Research Institute (EPRI) Materials Reliability Program (MRP). The MRP inspection and evaluation (I&E) guidelines for managing the effects of aging on pressurized water reactor vessel internals are presented in MRP-227-A, "Materials Reliability Program: Pressurized Water Reactor Internals Inspection and Evaluation Guidelines." The MRP also developed inspection requirements specific to the inspection methods delineated in MRP-227-A, as well as requirements for qualification of the nondestructive examination (NDE) systems used to perform those inspections. These inspection requirements are presented in MRP-228, "Materials Reliability Program: Inspection Standard for PWR Internals."

MRP-227-A and MRP-228 provide the basis of the IPEC Reactor Vessel Internals (RVI) Program. The RVI Program will monitor the effects of aging degradation mechanisms on the intended function of the internals through periodic and conditional examinations. The RVI Program will detect and evaluate cracking, loss of material, reduction of fracture toughness, loss of preload and dimensional changes of vessel internals components in accordance with MRP-227-A inspection requirements and evaluation acceptance criteria.

The IPEC RVI Program will be implemented and maintained in accordance with the guidance in NEI 03-08 [Addenda], Addendum A, "RCS Materials Degradation Management Program Guidelines." Any deviations from mandatory, needed, or good practice implementation requirements established in MRP-227-A or MRP-228, will be resolved in accordance with the NEI 03-08 implementation protocol. The RVI Program will be implemented prior to the period of extended operation.

## **B.1.42 Reactor Vessel Internals Program**

### **Program Description**

The Reactor Vessel Internals (RVI) Program is a new plant-specific program. Revision 1 of NUREG-1801 (Reference B.2-1) includes no aging management program description for PWR reactor vessel internals. NUREG-1801, Section XI.M16, PWR Vessel Internals, instead defers to the guidance provided in Chapter IV line items as appropriate. The Chapter IV line item guidance recommends actions to:

“... (1) participate in the 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; and (3) upon completion of these programs, but not less than 24 months before entering the period of extended operation, submit an inspection plan for reactor internals to the NRC for review and approval.”

The industry programs for investigating and managing aging effects on reactor internals are part of the Electric Power Research Institute (EPRI) Materials Reliability Program (MRP). The MRP developed inspection and evaluation (I&E) guidelines for managing the effects of aging on pressurized water reactor vessel internals. These guidelines, as reviewed and accepted by the NRC (Reference B.2-2), are presented in MRP-227-A, “Materials Reliability Program: Pressurized Water Reactor Internals Inspection and Evaluation Guidelines.” The I&E guidelines include:

- summary descriptions of PWR internals and functions;
- summary of the categorization and aging management strategy development of potentially susceptible locations, based on the safety and economic consequences of aging degradation;
- direction for methods, extent, and frequency of one-time, periodic, and conditional examinations and other aging management methodologies;
- acceptance criteria for the one-time, periodic, and conditional examinations and other aging management methodologies; and
- methods for evaluation of conditions that fail to meet the examination acceptance criteria.

The MRP also developed inspection procedure requirements specific to the inspection methods delineated in MRP-227-A, as well as requirements for qualification of the nondestructive examination (NDE) systems used to perform those inspections. These inspection procedure requirements are presented in MRP-228, “Materials Reliability Program: Inspection Standard for PWR Internals.”

Revision 2 of NUREG-1801 (Reference B.2-3) contains a new aging management program (XI.M16A) addressing PWR RVIs. This new aging management program relies on the implementation of EPRI report MRP-227, and applies the guidance in that document. In 2013, the NRC Staff issued Final License Renewal Interim Staff Guidance LR-ISG-2011-04, Updated Aging Management Criteria for Reactor Vessel Internal Components for Pressurized Water Reactors (Reference B.2-4), which revised the recommendations in NUREG-1801, Revision 2

and the NRC Staff's acceptance criteria and review procedures to ensure consistency with MRP-227-A and provide a framework to adequately address age-related degradation and aging management of RVI components during the period of extended operation.

MRP-227-A and MRP-228 provide the basis of the IPEC Reactor Vessel Internals (RVI) Program. Revisions to MRP-227-A and MRP-228 will be incorporated into the IPEC RVI Program.

The RVI Program will monitor the effects of aging on the intended function of the reactor vessel internals through periodic and conditional examinations. The RVI Program will detect and evaluate cracking, loss of material, reduction of fracture toughness, loss of preload and dimensional changes of vessel internals components in accordance with MRP-227-A inspection recommendations and evaluation acceptance criteria.

IPEC will implement and maintain the RVI Program in accordance with the guidance in NEI 03-08 [Addenda], Addendum A, "RCS Materials Degradation Management Program Guidelines." Any deviations from mandatory, needed, or good practice implementation activities established in MRP-227-A or MRP-228, will be managed in accordance with the NEI 03-08 implementation protocol.

The Reactor Vessel Internals Program is implemented through the Indian Point Energy Center Reactor Vessel Internals Inspection Plan (Reference B.2-53). The inspection plan provides additional details, including:

- Identification of items for inspection,
- Specification of the type of examination appropriate for each degradation mechanism,
- Specification of the required level of examination qualification,
- Schedule of initial inspection, schedule and frequency of subsequent inspections,
- Criteria for sampling and coverage,
- Criteria for expansion of scope if unacceptable indications are found,
- Inspection acceptance criteria,
- Methods for evaluating examination results not meeting the acceptance criteria,
- Provisions for updating the program based on industry-wide results, and
- Contingency measures to repair, replace or mitigate unacceptable examination results.

The Indian Point Energy Center Reactor Vessel Internals Inspection Plan also includes responses to applicable license renewal applicant action items identified in the NRC's safety evaluation of MRP-227 (incorporated in MRP-227-A).

## **Evaluation**

### **1. Scope of Program**

MRP-227-A guidelines are applicable to reactor internal structural components. The scope does not include consumable items such as fuel assemblies and reactivity control



assemblies which are periodically replaced based on neutron flux exposure. The scope does not include welded attachments to the reactor vessel which are considered part of the vessel, or nuclear instrumentation (flux thimble tubes) which forms part of the reactor coolant pressure boundary. Other programs manage the effects of aging on these components.

MRP-227-A separates PWR internals components into four groups depending on (1) their susceptibility to and tolerance of aging effects, and (2) the existence of programs that manage the effects of aging. These groupings include:

- Primary – those internals components that are highly susceptible to the effects of at least one aging mechanism (identified in Table 4-3 of MRP-227-A);
- Expansion – those internals components that are highly or moderately susceptible to the effects of at least one aging mechanism, but for which functionality assessment has shown a degree of tolerance to those effects (identified in Table 4-6 of MRP-227-A);
- Existing Programs – those internals components that are susceptible to the effects of at least one aging mechanism and for which generic and plant-specific existing AMP elements are capable of managing those effects (identified in Table 4-9 of MRP-227-A); and
- No Additional Measures – those internals components for which the effects of aging mechanisms are below the MRP-227-A screening criteria (internals components not included in Tables 4-3, 4-6 or 4-9 of MRP-227-A).

The categorization of internals components for Westinghouse PWRs, as presented in MRP-227-A, applies to IPEC Unit 2 and Unit 3 vessel internals. The component inspections identified in MRP-227-A, Tables 4-3 and 4-6 for primary and expansion group components, define the scope of the IPEC RVI Program inspections. Those components subject to aging management by existing programs, as delineated in MRP-227-A, Table 4-9, are included in the scope of those programs, and are not part of the RVI Program inspections. Components that are not included in Tables 4-3, 4-6 or 4-9 are considered to be within the scope of the program, but require no specific inspections.

## **2. Preventive Actions**

The Reactor Vessel Internals Program is a condition monitoring program that does not include preventive actions. However, primary water chemistry is maintained in accordance with EPRI guidelines by the Water Chemistry Control – Primary and Secondary Program, which minimizes the potential for loss of material, stress corrosion cracking (SCC), primary water stress corrosion cracking (PWSCC), and irradiation assisted stress corrosion cracking (IASCC).

Plant operations also influence aging of the vessel internals. The general assumptions about plant operations used in the development of the MRP-227-A guidelines are applicable to the IPEC units. Both units are base loaded and both implemented low leakage core loading patterns within the first 30 years of operation. IPEC has implemented no design changes to reactor vessel internals beyond those identified in general industry guidance or recommended by Westinghouse.

### **3. Parameters Monitored or Inspected**

The RVI Program will monitor the effects of aging on the intended function of the internals through periodic and conditional examinations and other aging management methods, as required. As described in MRP-227-A, the program contains elements that will monitor and inspect for the parameters that indicate the progress of each of these effects. The component inspections identified in MRP-227-A, Tables 4-3 and 4-6 for primary and expansion group components respectively, set forth the parameters monitored by the IPEC RVI Program inspections.

The program will use NDE techniques to detect loss of material through wear, identify changes in dimension due to void swelling and irradiation growth, distortion, or deflection, and locate cracks induced by SCC, PWSCC, IASCC, or fatigue/cyclical loading. Loss of preload, caused by thermal and irradiation-enhanced stress relaxation or creep, is indirectly monitored by inspecting for gross surface conditions that may be indicative of loosening in applicable bolted, fastened, keyed, or pinned connections. The reduction of fracture toughness, induced by either thermal aging or neutron irradiation embrittlement, is indirectly monitored by using visual or volumetric examination techniques to monitor for cracking in the components and by applying applicable reduced fracture toughness properties in flaw evaluations where warranted.

Visual examinations (VT-3) will be used to detect wear. Visual examinations (VT-3) will also detect distortion or cracking through indications such as gaps or displacement along component joints and broken or damaged bolt locking systems. Direct measurements of spring height will be used to detect distortion of the internals hold down spring. Visual examinations (EVT-1) will be used to detect broken components and crack-like surface flaws of components and welds. Volumetric (ultrasonic) examinations will be used to locate cracking of bolting.

### **4. Detection of Aging Effects**

The RVI Program will detect cracking, loss of material, reduction of fracture toughness, loss of preload and dimensional changes (distortion) of vessel internals components in accordance with the specific provisions of MRP-227-A. The NDE systems (i.e., the combinations of equipment, procedure, and personnel) used to detect these aging effects will be qualified in accordance with MRP-228. The RVI Program will conduct inspections of primary group components as delineated in MRP-227-A, Table 4-3. Indications from EVT-1 or UT inspections may result in additional inspections of expansion group components, as determined by expansion criteria delineated in MRP-227-A, Table 5-3. The relationships between primary group component inspection findings and additional inspections of expansion group components are as follows described in MRP-227A, Table 4-6.

### **5. Monitoring and Trending**

The RVI Program uses the inspection guidelines for PWR internals in MRP-227-A. Inspections in accordance with these guidelines will provide timely detection of aging effects. In addition to the inspections of primary group components, expansion group components have been defined should the scope of examination and re-examination require

expansion beyond the primary group. Records of inspection results are maintained allowing for comparison with subsequent inspection results.

In accordance with MRP-227-A, IPEC will provide a summary report of all inspections and monitoring, items requiring evaluation, and new repairs to the MRP Program Manager. The IPEC-specific results will be incorporated into an overall industry report that will track industry progress and will aid in evaluation of potentially significant issues, identification of fleet trends, and determination of any needed revisions to the MRP-227-A guidelines.

## **6. Acceptance Criteria**

The RVI Program acceptance criteria are from provided in Section 5 of MRP-227-A. Table 5-3 and Sections 5.1 through 5.3 of MRP-227-A provide the acceptance criteria for inspections of the IPEC primary and expansion group components. The criteria for expanding the examinations from the primary group components to include the expansion group components are also delineated in MRP-227-A, Table 5-3. The examination acceptance criteria include: (i) specific, descriptive relevant conditions for the visual (VT-3) examinations; (ii) requirements for recording and dispositioning surface breaking indications that are detected and sized for length by the visual (EVT-1) examinations; (iii) requirements for system-level assessment of bolted assemblies with unacceptable volumetric (UT) examination indications that exceed specified limits and (iv) requirements for fit up limits on physical measurements of the hold down springs.

## **7. Corrective Action**

Conditions adverse to quality, such as failures, malfunctions, deviations, defective material or equipment, and nonconformances, are promptly identified and corrected. In the case of significant conditions adverse to quality, measures are implemented to ensure that the cause of the nonconformance is determined and that corrective action is taken to preclude recurrence. In addition, the cause of the significant condition adverse to quality and the corrective action implemented is documented and reported to appropriate levels of management. The Entergy (10 CFR Part 50, Appendix B) Quality Assurance Program, including relevant corrective action controls, applies to the RVI Program.

Any detected condition that fails to meet the examination acceptance criteria must be processed through the corrective action program. Example methods for analytical disposition of unacceptable conditions are discussed or referenced in Section 6 of MRP-227-A. The evaluation methods include recommendations for flaw depth sizing and for crack growth determinations as well for performing applicable limit load, linear elastic and elastic-plastic fracture analyses of relevant flaw indications. These methods or other NRC-approved evaluation methods may be used. The alternative of component repair and replacement of PWR vessel internals is subject to the applicable requirements of the ASME Code Section XI.

## **8. Confirmation Process**

This attribute is discussed in Section B.0.3.

## **9. Administrative Controls**

This attribute is discussed in Section B.0.3.

## 10. Operating Experience

From an overall fleet-wide perspective, relatively few incidents of PWR internals aging degradation have been reported in operating U.S. commercial PWR plants. However, PWR internals aging degradation has been observed in European PWRs, (and, more recently, in certain U.S. PWRs) specifically with regard to cracking of baffle-former bolting. For this reason, the U.S. PWR owners and operators created a program to inspect the baffle-former bolting to determine whether similar aging degradation might be expected to occur in U.S. plants. A benefit of this decision was the experience gained with the UT examination techniques used in the inspections.

Since the Spring of 2016, baffle-former bolt degradation has been detected at some operating plants in the United States. As a result of the baffle-former bolt inspection findings, Westinghouse issued Nuclear Safety Advisory Letter NSAL-16-1, which contains a description of the issue, a technical evaluation, and recommended actions for utilities to follow. This NSAL recommended that tier 1a plants (i.e. Westinghouse, 4-loop, downflow plants with 347 stainless steel bolting) similar to the IPEC Units should perform volumetric examinations of the baffle-former bolts at the next refueling outage. As a result, IP3 moved the inspections from the 2019 outage to the 2017 outage to comply with the NSAL recommendations as well as the Interim Guidance issued by the EPRI MRP Program in MRP 2016-022, *Transmittal of NEI-03-08 "Needed" Interim Guidance Regarding Baffle Former Bolt inspections for Tier 1 plants as Defined in Westinghouse NSAL 16-01.*

In the spring of 2016, during IP2 outage 2R22, ultrasonic (UT) and/or visual inspections of all 832 baffle-former bolts (bolts) were performed in accordance with the NRC approved guidelines in MRP-227-A. Visual inspection of the baffle plates and bolts identified 31 degraded bolts. The UT inspections identified indications on 182 bolts and also determined that 14 bolt locations were not testable. The locations that were not testable were conservatively assumed to possess bolts that failed to meet the acceptance criteria. As a result of the inspection findings, all 227 bolts (31+182+14) with actual and assumed indications were replaced. An additional 51 bolts were replaced to reduce the probability of future failures as well as minimize the probability of clusters of failed bolts. Therefore, during 2R22, a total of 278 bolts (227+51) were replaced.

As a result of the IP2 inspection findings and other industry Operating Experience (OE) indicating a significant number of failed bolts at other similarly-designed PWR plants, the IPEC PWR Vessel Internals Program, SEP-PVI-IPEC-001 was revised. In view of the 2R22 inspection findings, Entergy arranged for the fractographic examination of eight baffle-former bolts removed from the IP2 baffle structure during the Spring 2016 outage at Westinghouse Electric Company's hot cell laboratory in Churchill, PA. The results of those fractographic examinations are documented in Westinghouse Report MCOE-TR-16-18, Revision 0, "Fractography of Indian Point Unit 2 Baffle Former Bolts" (Nov. 30, 2016). Industry-sponsored metallurgical analysis and materials property testing of additional baffle former bolt specimens from IP2 and other PWRs is still in progress.

In addition, the industry undertook laboratory testing projects to gather the materials data necessary to support future inspections and evaluations. Other confirmed or suspected material degradation concerns that the industry has identified for PWR components are wear in thimble tubes, potential wear in control rod guide tube guide plates, and cracking in some high-strength

bolting. The industry has addressed the last concern primarily through replacement of high-strength bolting with bolt material that is less susceptible to cracking and by improved control of pre-load.

The RVI Program established in accordance with the MRP-227-A guidelines is a new program. Accordingly, there is no direct programmatic history for IPEC. However, program inspections will use qualified techniques similar to those successfully used at IPEC and throughout the industry for ASME Section XI Code inspections. Internals inspections (VT-3) have been conducted at IPEC in accordance with ASME Section XI Code requirements, with no indications of component degradation. IPEC has appropriately responded to industry operating experience for reactor vessel internals. For example, guide tube support pins (split pins) have been replaced in both units on the basis of industry experience. As with other U.S. commercial PWR plants, cracking of baffle former bolts is recognized as a potential issue for the IPEC units. As a result, IPEC has monitored industry developments and recommendations regarding these components.

Development of the MRP-227-A guidelines is based upon industry operating experience, research data, and vendor evaluations. Reactor vessels internals aging degradation incidents in both U.S. and foreign plants were considered in the development of the MRP- 227-A guidelines. As implemented, this program will account for applicable future operating experience during the period of extended operation.

### **Conclusion**

The RVI Program will be effective at managing aging effects since it will incorporate proven monitoring techniques, acceptance criteria, corrective actions, and administrative controls in accordance with MRP-227-A and MRP-228 guidelines and current IPEC programs. The RVI Program will provide reasonable assurance that the effects of aging are managed such that applicable components will continue to perform their intended functions consistent with the current licensing basis through the period of extended operation.

## B.2 REFERENCES

~~B.2-1~~ NUREG-1800, Standard Review Plan for Review of License Renewal Applications for Nuclear Power Plants, U.S. Nuclear Regulatory Commission, September 2005.

B.2-12 NUREG-1801, Revision 1, Generic Aging Lessons Learned (GALL) Report, U.S. Nuclear Regulatory Commission, September 2005.

B.2-2 Letter from R. Nelson, U.S. Nuclear Regulatory Commission, to N. Wilmshurst, Electric Power Research Institute, Final Safety Evolution of EPRI Report, Materials Reliability Program Report 1016596 (MRP-227), Revision 0, Pressurized Water Reactor (PWR) Internals Inspection and Evaluation Guidelines (TAC No. ME0680), June 2011

B.2-3 NUREG-1801, Revision 2 Generic Aging Lessons Learned (GALL) Report, U.S. Nuclear Regulatory Commission, December 2010

B.2-4 Final Interim Staff Guidance LR-ISG-2011-04; Updated Aging Management Criteria for Reactor Vessel Internal Components for Pressurized Water Reactors, 78 Fed. Reg. 33,120 (June 3, 2013)

B.2-53 Indian Point Energy Center Reactor Vessel Internals Inspection Plan.

**ATTACHMENT 2 TO NL-17-020**

**INDIAN POINT ENERGY CENTER**

**REVISED REACTOR VESSEL INTERNALS INSPECTION PLAN**

Additions Underlined  
Deletions Lined Out

**ENTERGY NUCLEAR OPERATIONS, INC.  
INDIAN POINT NUCLEAR GENERATING UNIT NOS. 2 AND 3  
DOCKET NOS. 50-247 AND 50-286**

*Indian Point Energy Center  
Reactor Vessel Internals Inspection Plan*

# 1 INTRODUCTION

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## 1.1 Aging Management Program Inspection Plan

The EPRI MRP guidelines define a supplemental inspection program for managing aging effects on the reactor vessel internals and were used to develop this inspection plan for IPEC Units 2 and 3. The EPRI MRP Reactor Internals Focus Group developed the MRP-227-A guidelines to support the demonstration of continued functionality, with requirements for inspections to detect the effects of aging along with requirements for the evaluation of detected aging effects, if any. The development of MRP-227-A combined the results of component functionality assessments with component accessibility, operating experience, existing evaluations and prior examination results to determine the appropriate aging management methods, initial examination timing and the need and timing for subsequent inspections and identified the components and locations for supplemental examination.

This inspection plan includes:

- Identification of items for inspection,
- Specification of the type of examination appropriate for each degradation mechanism,
- Specification of the required level of examination qualification,
- Schedule of initial inspection schedule and frequency of subsequent inspections,
- Criteria for sampling and coverage,
- Criteria for expansion of scope if unacceptable indications are found,
- Inspection acceptance criteria,
- Methods for evaluating examination results not meeting the acceptance criteria,
- Provisions for updating the program based on industry-wide results; and
- Contingency measures to repair, replace or mitigate unacceptable examination results.



*Indian Point Energy Center  
Reactor Vessel Internals Inspection Plan*

# 2

## BACKGROUND OF IPEC REACTOR VESSEL INTERNALS DESIGN

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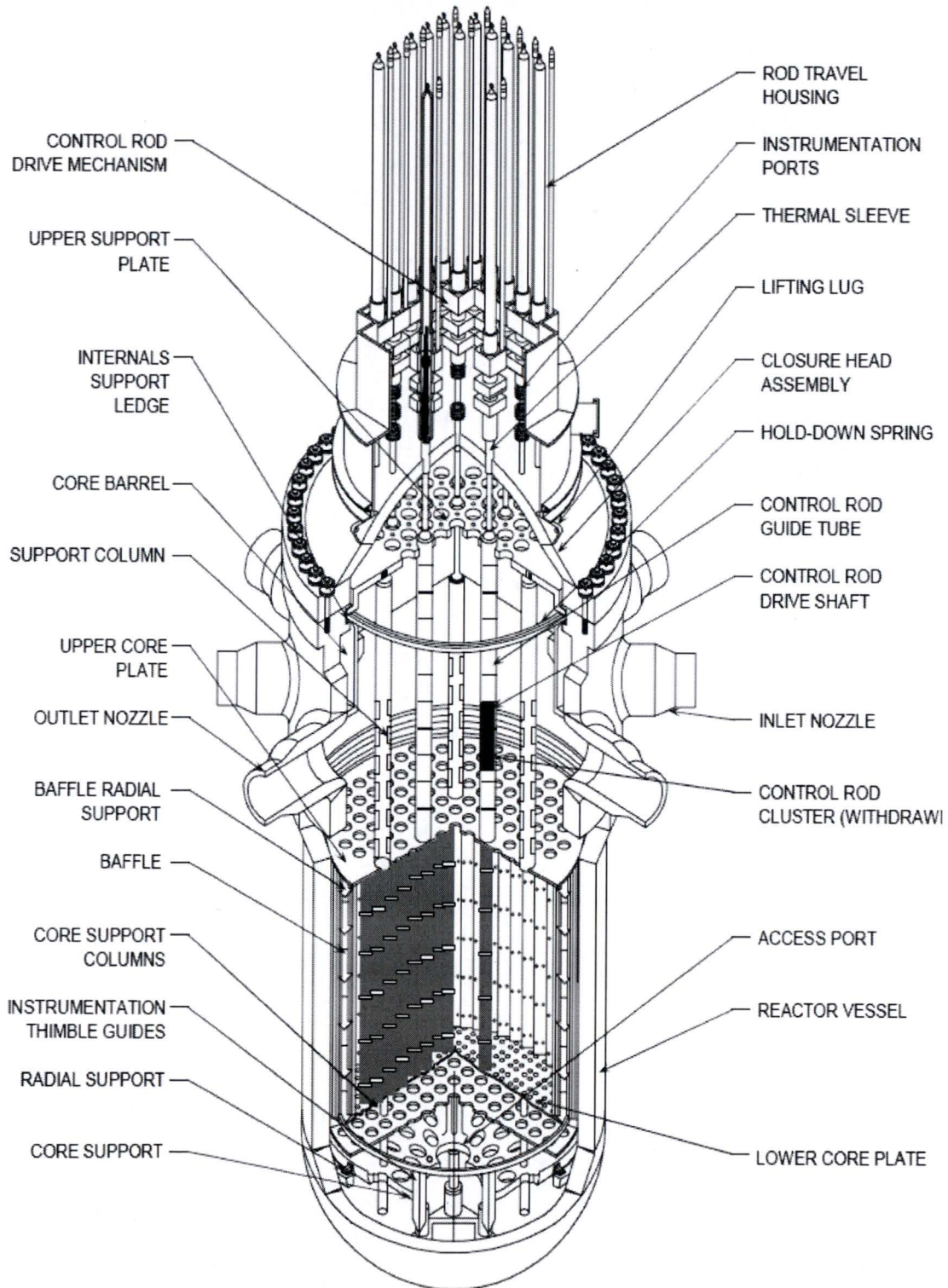
This section provides a summary of the design characteristics for the IPEC Westinghouse PWR internals.

### ***2.1 Westinghouse Internals Design Characteristics***

A schematic view of a typical set of Westinghouse-designed PWR internals is Figure 2-1. More detailed views of selected internals components are Figures 2-2 through 2-16 at the end of this section. These figures are typical and are not an exact representation of the IPEC internals.

To help in the categorization of IPEC internals design characteristics as discussed in MRP-227-A Section 3.1.3, the following information is provided. IPEC Units 2 and 3 are Westinghouse four loop plants with a downflow baffle-barrel region flow design, and a top hat design upper support plate. Unit 2 had an original thermal output rating of 2758 MWth and Unit 3 had an original thermal output rating of 3025 MWth. Unit 2 has a current thermal output rating of 3216 MWth and Unit 3 has a current thermal output rating of 3188 MWth.

*Indian Point Energy Center  
Reactor Vessel Internals Inspection Plan*



**Figure 2-1**  
**Overview of typical Westinghouse internals**

*Indian Point Energy Center  
Reactor Vessel Internals Inspection Plan*

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 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 and the lower core plate. 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 upper core plate 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 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.

### **Upper Internals Assembly**

The major sub-assemblies that constitute the upper internals assembly are the: (1) upper core plate (UCP); (2) upper support column assemblies; (3) control rod guide tube assemblies; and (4) upper support plate (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 USP acts as the divider between the upper plenum and the reactor vessel head and as a relatively stiff base for the rest of the upper internals. 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 IPEC is designated as a top hat design.

The UCP is perforated to permit coolant to pass from the core below into the upper plenum between 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 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 lower core plate (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 so they can be easily removed if replacement is desired. The control rod guide tubes are designed to guide the control rods in and out of the fuel assemblies to

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control power generation. Guide tube cards are located within each control rod guide tube 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 upper instrumentation columns are bolted to the USP. These columns support the thermocouple guide tubes that lead the thermocouples from the reactor head through the upper plenum to just above the UCP.

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. The function of the lower support casting is to provide support for the core. 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.

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 shields, 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 IPEC, 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.

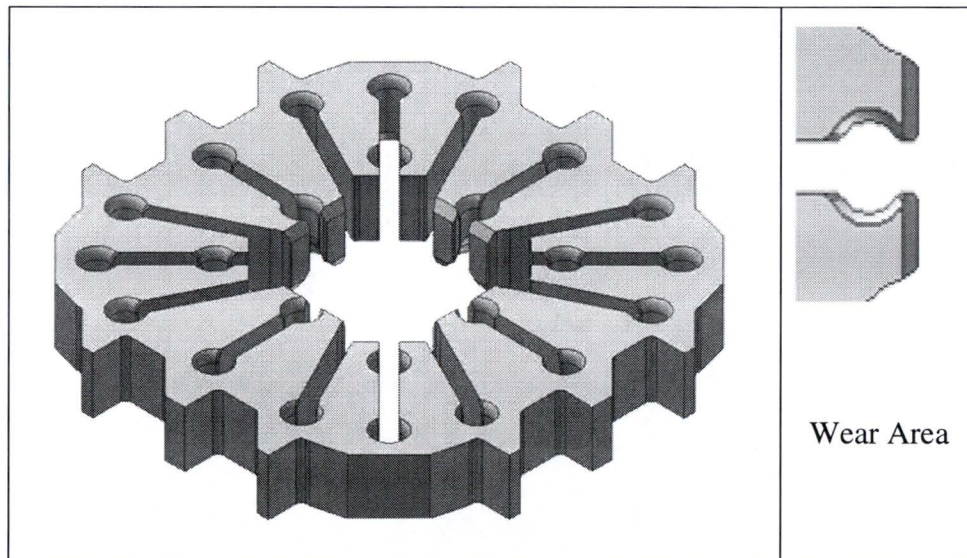
The function of the core barrel outlet nozzles is to direct the reactor coolant, after it leaves the core, radially outward through the reactor vessel outlet nozzles. The core barrel outlet nozzles are located in the upper portion of the core barrel directly below the flange and are attached to the core barrel, each in line with a vessel outlet nozzle.

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Additional neutron shielding of the reactor vessel is provided in the active core region by thermal shields attached to the outside of the core barrel.

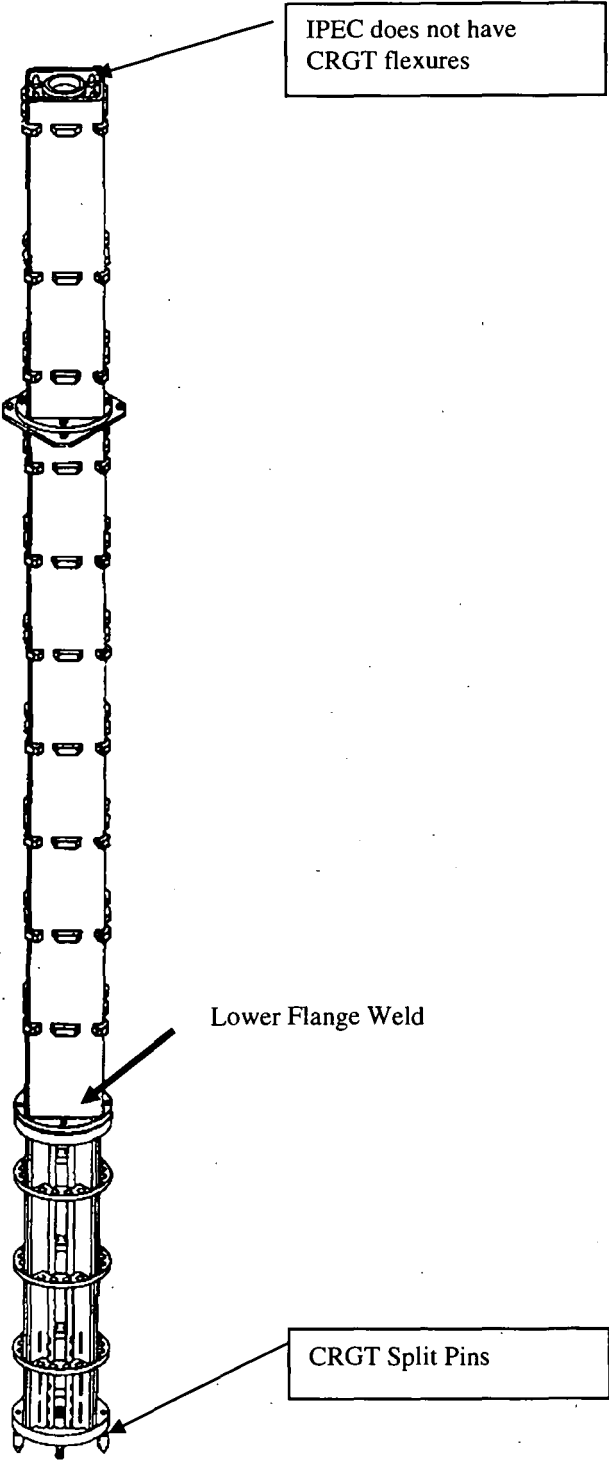
A flux thimble is a long, slender stainless steel tube that passes from an external seal table, through a bottom mounted nozzle penetration, through the lower internals assembly, and finally extends to the top of a fuel assembly. The flux thimble provides a path for a neutron flux detector into the core and is subjected to reactor coolant pressure and temperature on the outside surface and to atmospheric conditions on the inside. The flux thimble path from the seal table to the bottom mounted nozzles is defined by flux thimble guide tubes, which are part of the primary pressure boundary and not part of the internals. The bottom-mounted instrumentation (BMI) columns provide a path for the flux thimbles from the bottom of the vessel into the core. The BMI columns align the flux thimble paths with instrumentation thimbles in the fuel assembly.

In the upper internals assembly, the upper support plate, the upper support columns, and the upper core plate are considered core support structures. In the lower internals assembly the lower core plate, 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.



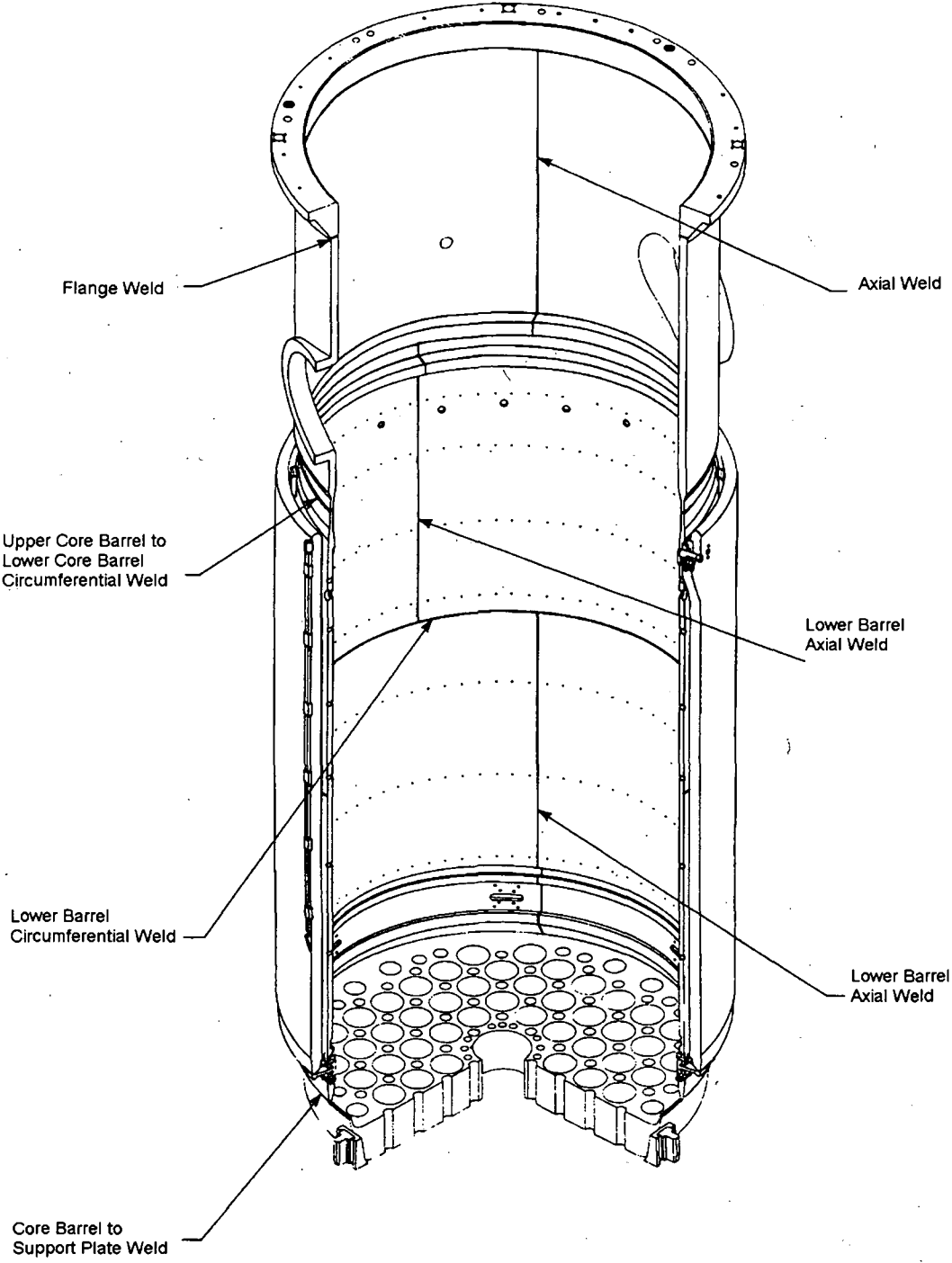
**Figure 2-2  
Typical Westinghouse control rod guide card (17x17 fuel assembly)**

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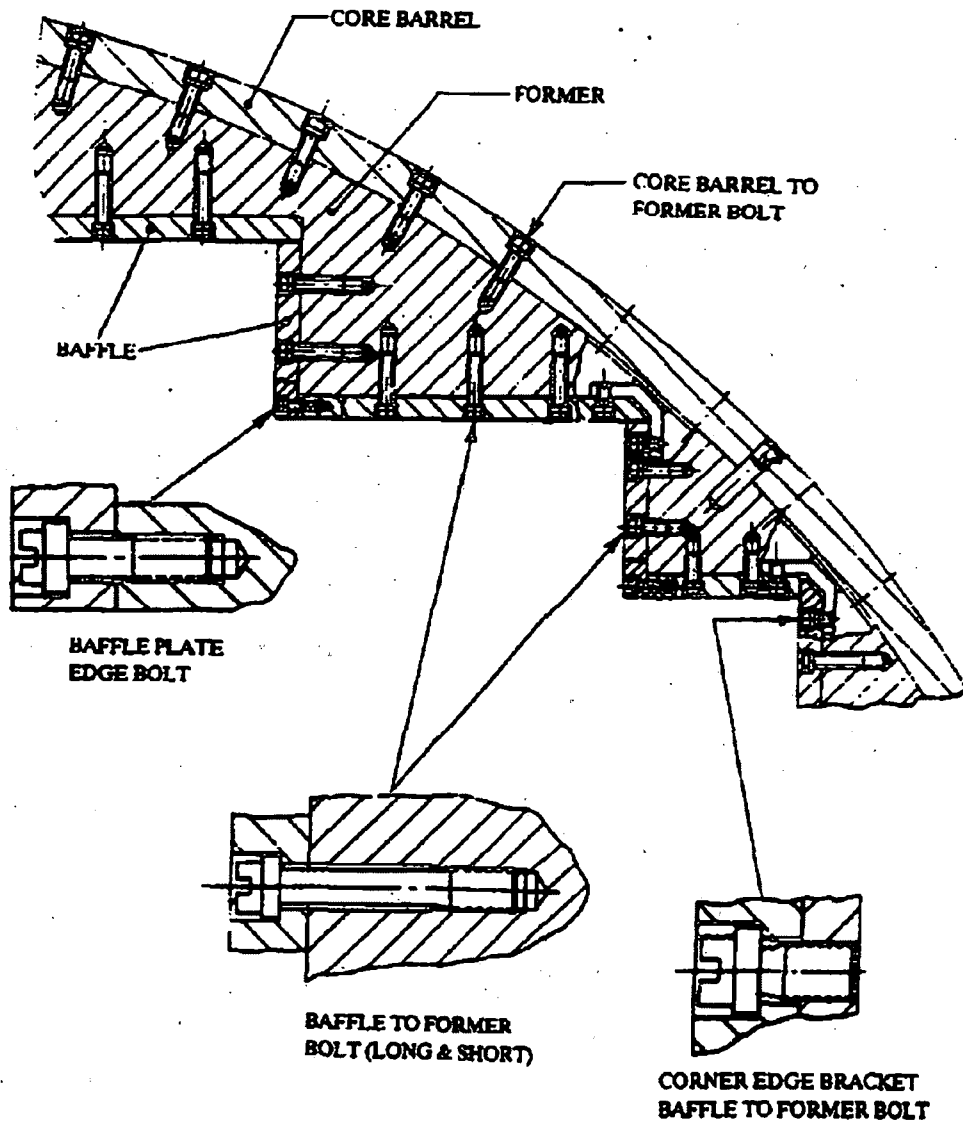
**Figure 2-3  
Typical Westinghouse control rod guide tube assembly**

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**Figure 2-4  
Major fabrication welds in typical Westinghouse core barrel**

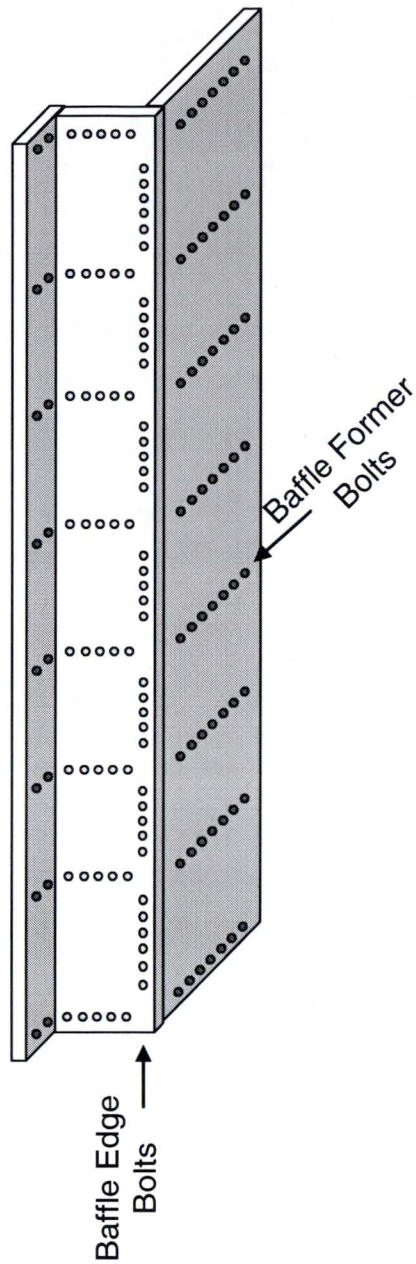
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**Figure 2-5**  
Bolt locations in typical Westinghouse baffle-former-barrel structure.

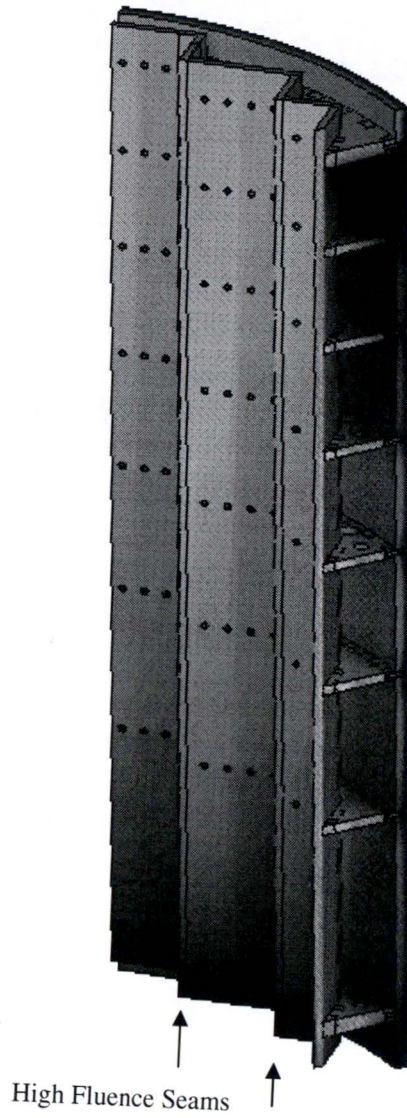


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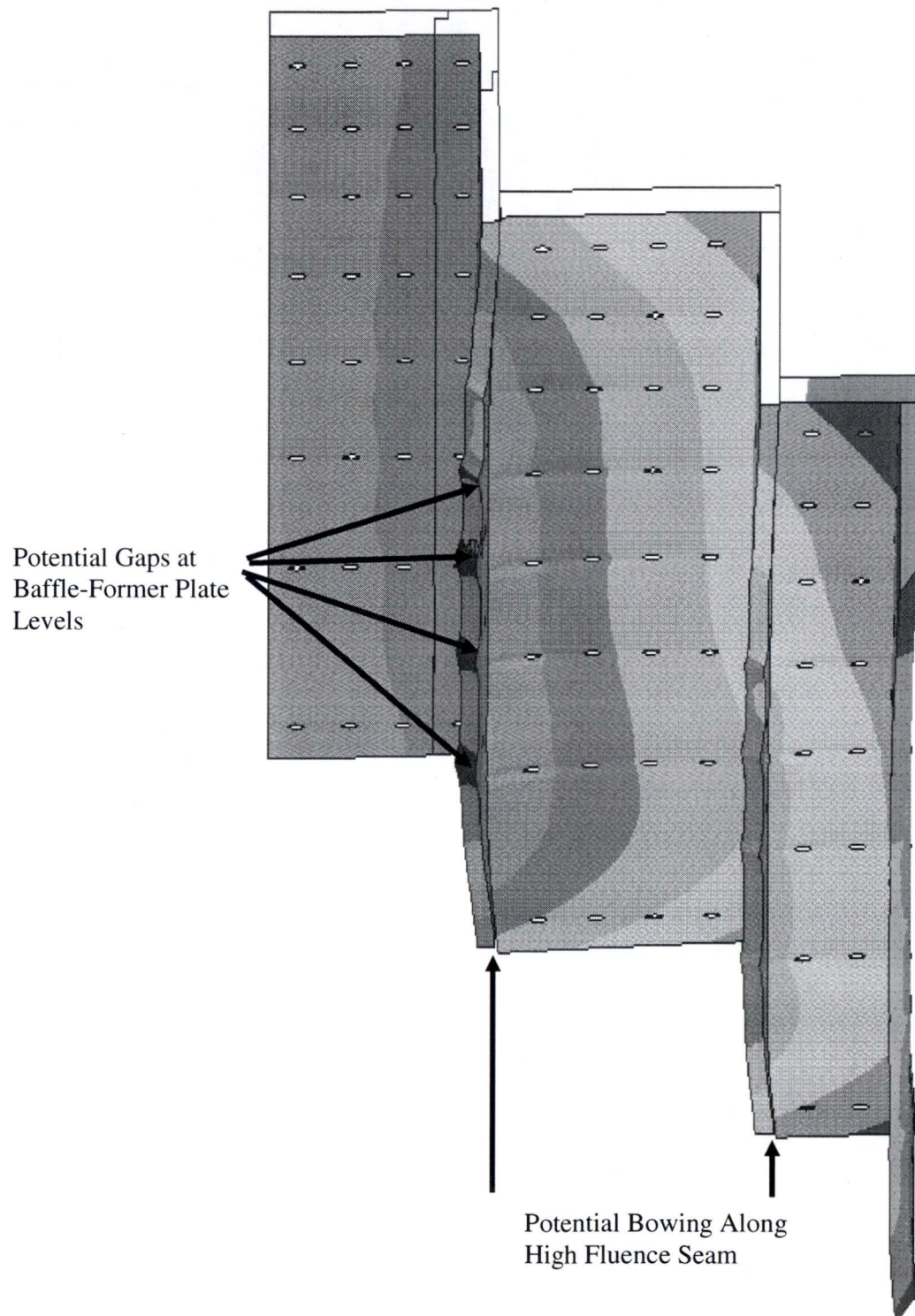
**Figure 2-6**  
**Baffle-edge bolt and baffle-former bolt locations at high fluence seams in bolted baffle-former assembly**

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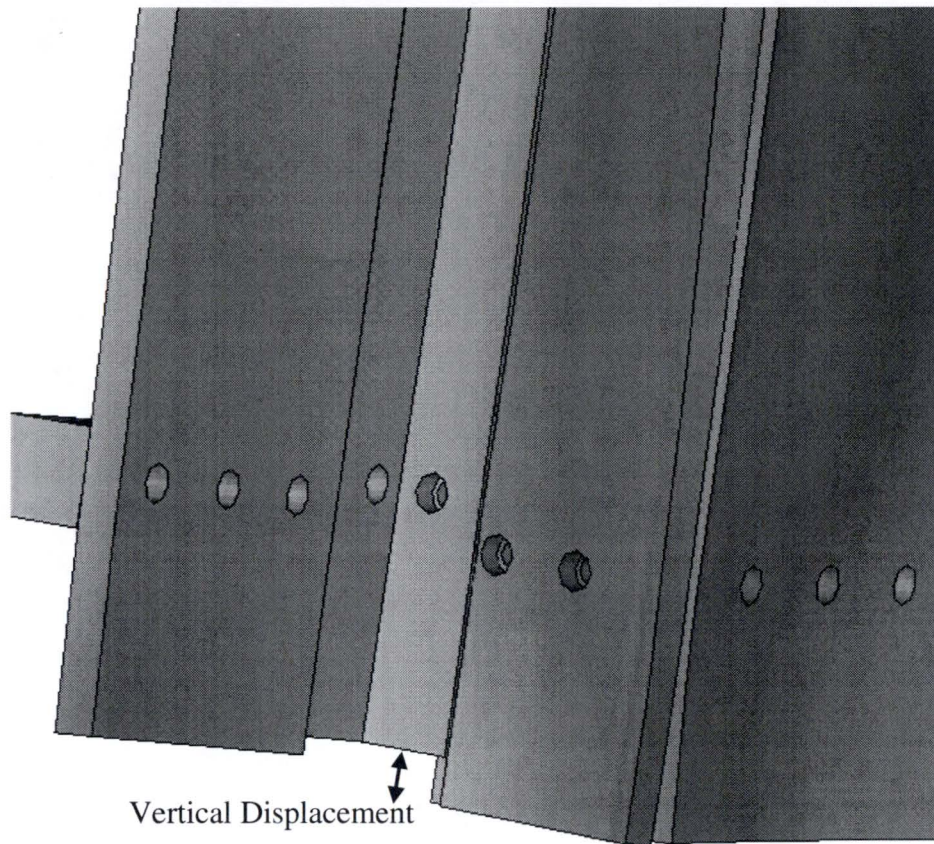
**Figure 2-7**  
**High fluence seam locations in Westinghouse baffle-former assembly**

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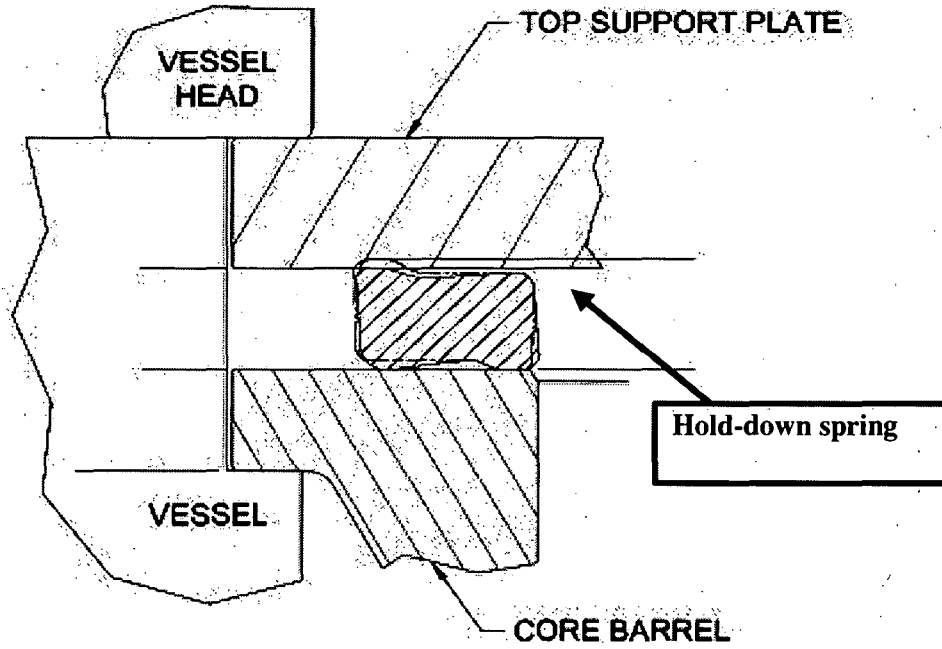
**Figure 2-8**  
**Exaggerated view of void swelling induced distortion in Westinghouse baffle-former assembly.**

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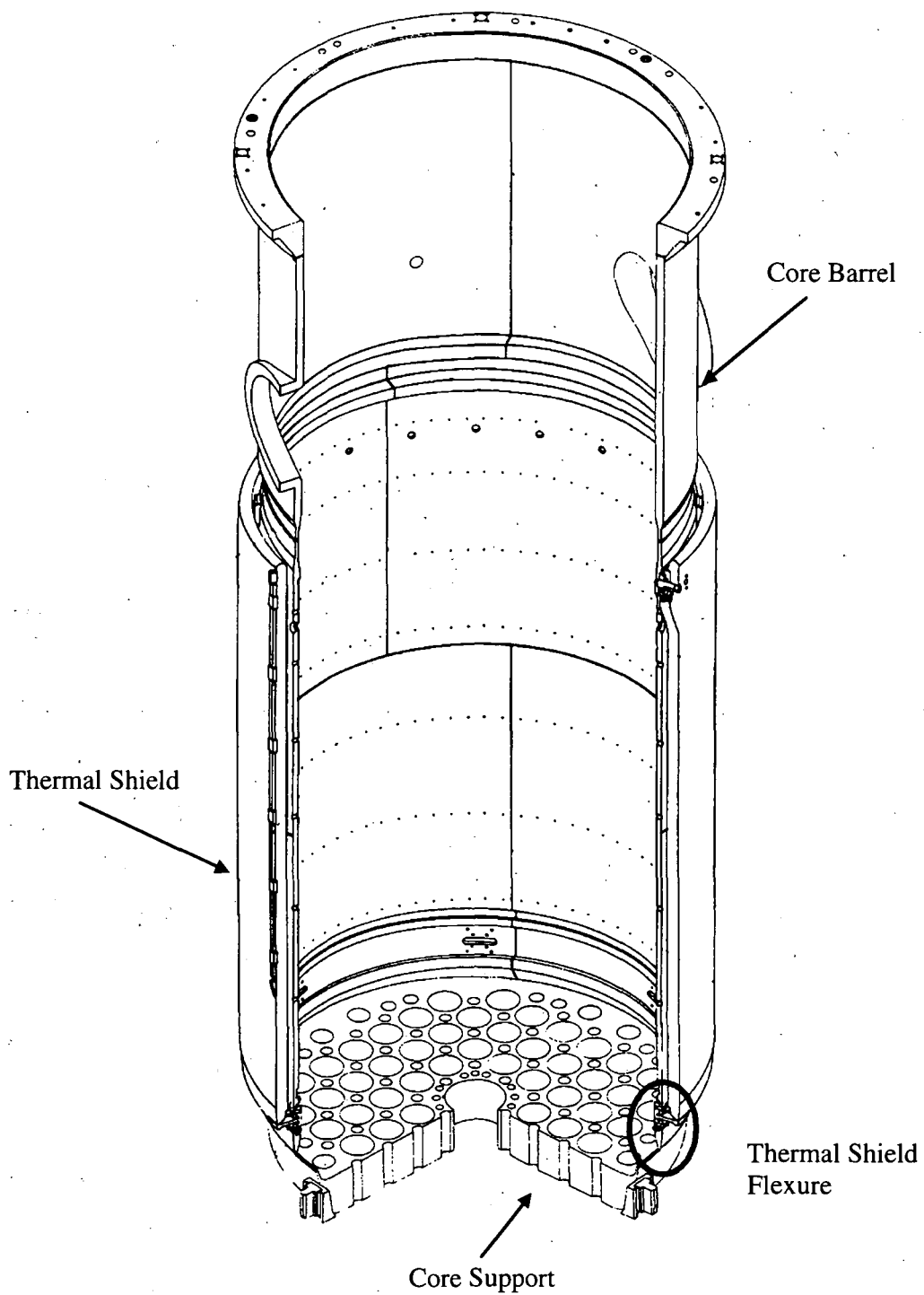
**Figure 2-9**  
Vertical displacement of Westinghouse baffle plates caused by void swelling.

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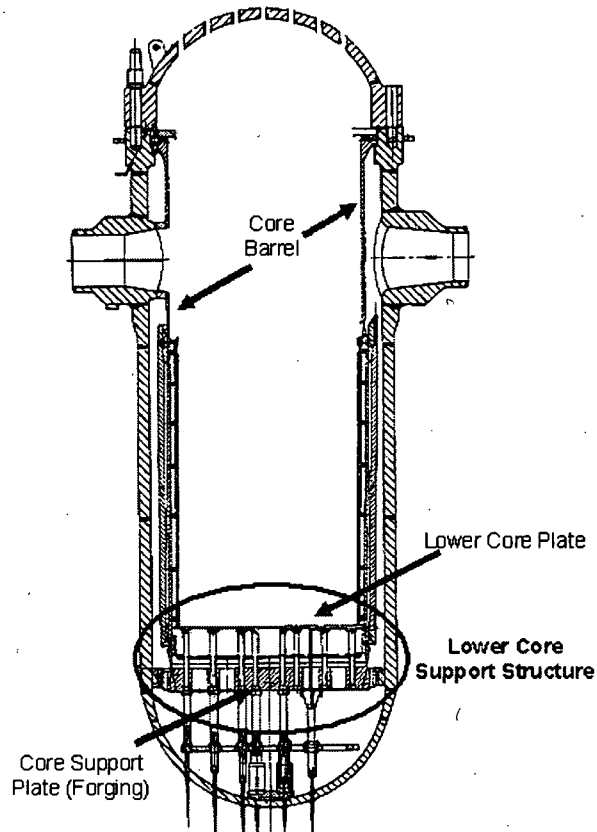
**Figure 2-10**  
**Schematic cross-sections of the Westinghouse hold-down springs**

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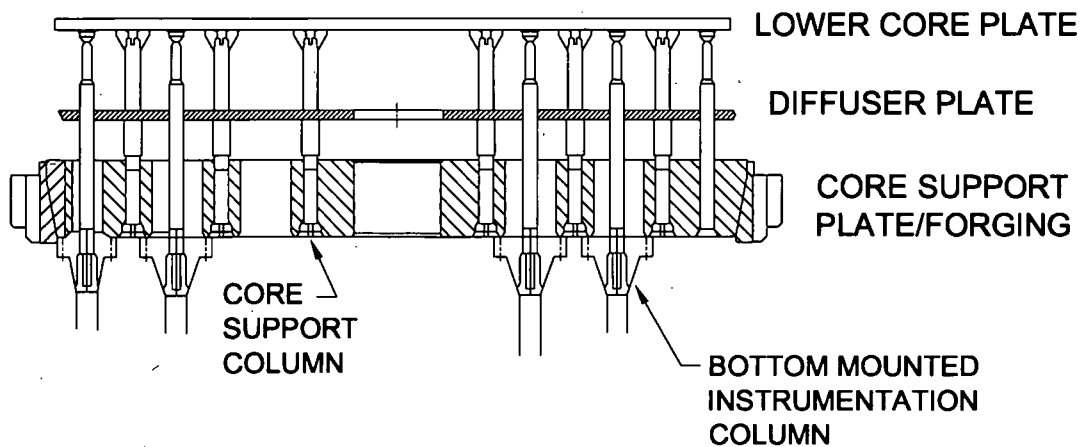


**Figure 2-11  
Location of Westinghouse thermal shield flexures**

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**Figure 2-12**  
Schematic indicating location of Westinghouse lower core support structure. Additional details shown in Figure 2-13

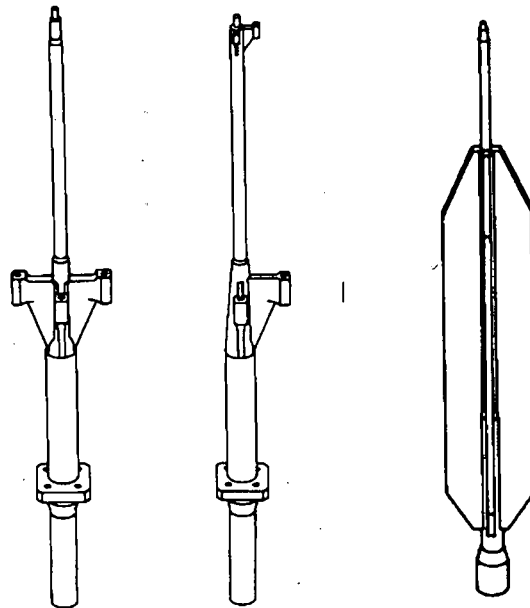


**Figure 2-13**  
Westinghouse lower core support structure and bottom mounted instrumentation columns. Core support column bolts fasten the core support columns to the lower core plate

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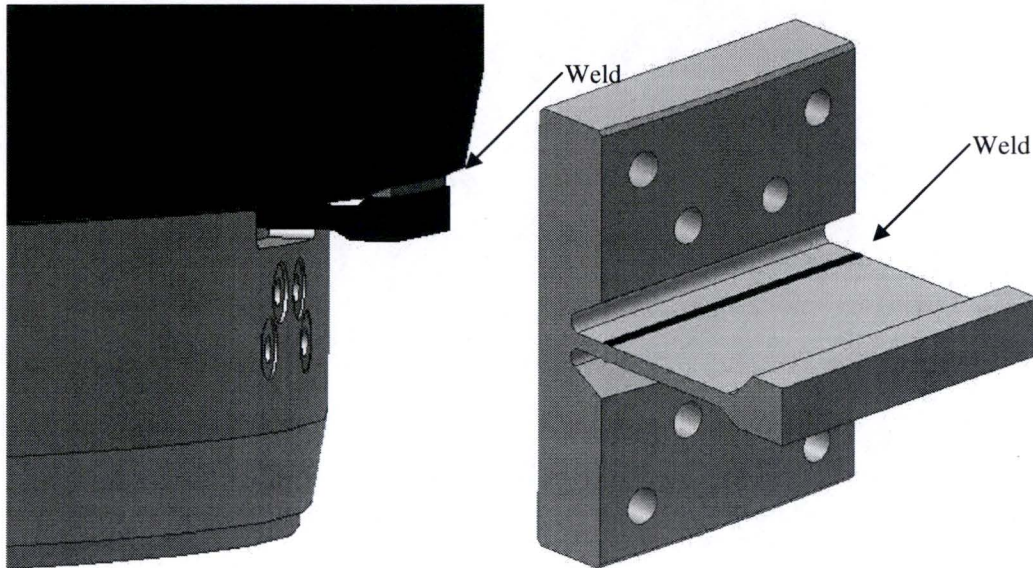
**Figure 2-14**  
Typical Westinghouse core support column. Core support column bolts fasten the top of the support column to the lower core plate



**Figure 2-15**  
Examples of Westinghouse bottom mounted instrumentation column designs



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**Figure 2-16**  
**Typical Westinghouse thermal shield flexure**

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# 3

## INSPECTION PLAN SUMMARY

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Management of component aging effects includes actions to prevent or control aging effects, review of operating experience to better understand the potential for aging effects to occur, inspections to detect the onset of aging effects in susceptible components, and protocols for evaluation and remediation of the effects of aging.

### 3.1 Component Inspection and Evaluation Overview

This discussion summarizes the guidance of the MRP Inspection & Evaluation (I&E) guidelines necessary to understand implementation but does not duplicate the full discussion of the technical bases. MRP-227-A and its supporting documents provide further information on the technical bases of the program.

MRP-227-A establishes four groups of reactor internals components with respect to inspections: Primary, Expansion, Existing Programs and No Additional Measures, as summarized below.

- **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 which 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 a 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 will depend on the findings from the examinations of the Primary components.
- **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:** 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. Items categorized as Category A in MRP-191 are those for which aging effects are below the screening criteria, so that aging degradation significance is minimal. Primary, expansion, and existing examinations verify that the chemical control program has been effective at controlling stress corrosion cracking and loss of material due to corrosion for Category A components.

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Additional components were placed in the No Additional Measures group as a result of Failure Modes, Effects and Criticality Analysis (FMECA) and the functionality assessment. No further action is required for managing the effects of aging of the No Additional Measures components. However, any core support structures subject to ASME Section XI Examination Category B-N-3 requirements continue to be subject to those ASME Code requirements throughout the period of extended operation.

The inspection methods required for Primary and Expansion components were selected from visual, surface and volumetric examination methods that are applicable and appropriate for the expected degradation effect (e.g. cracking caused by particular mechanisms, loss of material caused by wear). The inspection methods include: Visual examinations (VT-3, VT-1, EVT-1), surface examinations, volumetric examinations (specifically UT) and physical measurements. MRP-227-A provides detailed justification for the components selected for inspection and the specific examination methods selected for each. The MRP-228 report, PWR Internals Inspection Standards, provides detailed examination standards and any inspection technical justification or inspection personnel training requirements.

### **3.2 Inspection and Evaluation Requirements for Primary Components**

The inspection requirements for Primary Components at IPEC Units 2 and 3 from MRP-227-A are provided in Table 5-2.

### **3.3 Inspection and Evaluation Requirements for Expansion Components**

The inspection requirements for Expansion Components at IPEC Units 2 and 3 from MRP-227-A are provided in Table 5-3.

### **3.4 Inspections of Existing Program Components**

The list of Existing Program Components at IPEC Units 2 and 3 from MRP-227-A are provided in Table 5-4. This includes components in the Section XI ISI Program categories B-N-2 and B-N-3 for IPEC Units 2 and 3.

The Reactor Vessel Component Inspections conducted as part of the ISI Program for IPEC Units 2 and 3 are listed in Table 5-6. The ISI Program inspections are implemented in accordance with ASME Section XI schedule requirements.

### **3.5 Examination Systems**

Equipment, techniques, procedures and personnel used to perform examinations required under this program will be consistent with the requirements of MRP-228. Indications detected during

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these examinations will be characterized and reported in accordance with the requirements of MRP-228.

### **3.6 Information Supplied in Response to the NRC Safety Evaluation of MRP-227-A**

As part of the NRC Revision 1 to the Final Safety Evaluation of MRP-227, a number of action items and conditions were specified by the staff. Table 5-8 summarizes the IPEC response to the NRC Final Safety Evaluation of MRP-227. Topical Report Conditions from the NRC Final Safety Evaluation of MRP-227 have been addressed in MRP-227-A. These items have been addressed in the appropriate sections of this document. Applicant/Licensee Action Items from the NRC Final Safety Evaluation of MRP-227 are discussed in this section.

#### **SER Section 4.2.1, Applicant/Licensee Action Item 1**

IPEC has assessed its plant design and operating history and has determined that MRP-227-A is applicable to the facility. The assumptions regarding plant design and operating history made in MRP-191 are appropriate for IPEC and there are no differences in component inspection categories at IPEC. IPEC Unit 2 (IP2) had the first 8 years of operation with a high leakage core loading pattern. IPEC Unit 3 (IP3) had the first 10 years of operation with a high leakage core loading pattern. The FMECA and functionality analyses were based on the assumption of 30 years of operation with high leakage core loading patterns; therefore, IPEC is bounded by the assumptions in MRP-191. IPEC has always operated as a base-load plant which operates at fixed power levels and does not vary power on a calendar or load demand schedule.

#### **SER Section 4.2.2, Applicant/Licensee Action Item 2**

IPEC reviewed the information in Table 4-4 of MRP-191 and determined that this table contains all of the RVI components that are within the scope of license renewal. This is shown in Table 5-7.

#### **SER Section 4.2.3, Applicant/Licensee Action Item 3**

At IP2, the original X750 guide tube support pins (split pins) were replaced in 1995 (after 21 years in service) with an improved X750 Revision B material made from more selective material with more continuous carbide coverage grain boundaries and tighter quality controls, to provide greater resistance to stress corrosion cracking. IP2 plans to begin preliminary split pin replacement engineering and walkdowns in 2014 and replace the split pins in 2016.

#### **SER Section 4.2.4, Applicant/Licensee Action Item 4**

This action item does not apply to Westinghouse designed units.

#### **SER Section 4.2.5, Applicant/Licensee Action Item 5**

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The IPEC plant specific acceptance criteria for hold down springs and an explanation of how the proposed acceptance criteria are consistent with the IPEC licensing basis and the need to maintain the functionality of the hold down springs under all licensing basis conditions will be developed prior to the first required physical measurement. The acceptance criteria will ensure the remaining compressible height of the spring shall provide hold down forces within the IPEC design tolerance. If a plant specific acceptance criterion is not developed for the hold down spring, IPEC will replace the spring in lieu of performing the first required physical measurement.

**SER Section 4.2.6, Applicant/Licensee Action Item 6**

This action item does not apply to Westinghouse designed units.

**SER Section 4.2.7, Applicant/Licensee Action Item 7**

The IPEC plant specific analyses to demonstrate the lower support column bodies will maintain their functionality during the period of extended operation will consider the possible loss of fracture toughness in these components due to thermal and irradiation embrittlement. The analyses will be consistent with the IPEC licensing basis and the need to maintain the functionality of the lower support column bodies under all licensing basis conditions of operation. IPEC will submit this information to the NRC prior to the period of extended operation.

**SER Section 4.2.8, Applicant/Licensee Action Item 8**

A Reactor Vessel Internals AMP description for IPEC was included in Amendment 9 to the License Renewal Application (NL-10-063, July 14, 2010). The AMP description has been revised to be consistent with MRP-227-A. The revised AMP description has been submitted under letter NL-12-037.

This document comprises an inspection plan which addresses the identified plant-specific action items contained in the NRC Revision 1 to the Final Safety Evaluation for MRP-227. IPEC is not requesting any deviations from the guidance provided in MRP-227-A.

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# 4

## EXAMINATION ACCEPTANCE AND EXPANSION CRITERIA AND IMPLEMENTATION REQUIREMENTS

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### 4.1 Examination Acceptance Criteria

#### 4.1.1 Visual (VT-3) Examination

Visual (VT-3) examination is an appropriate NDE method for the detection of general degradation conditions in many of the susceptible components. The ASME Code Section XI, Examination Category B-N-3, provides a set of relevant conditions for the visual (VT-3) examination of removable core support structures in Section IWB. These are:

1. structural distortion or displacement of parts to the extent that component function may be impaired;
2. loose, missing, cracked, or fractured parts, bolting, or fasteners;
3. corrosion or erosion that reduces the nominal section thickness by more than 5%;
4. wear of mating surfaces that may lead to loss of function; and
5. structural degradation of interior attachments such that the original cross-sectional area is reduced more than 5%.

For components in the Existing Programs group, these general relevant conditions are sufficient. However, for components where visual (VT-3) is specified in the Primary or the Expansion group, more specific descriptions of the relevant conditions are provided in Table 5-5 for the benefit of the examiners. One or more of these specific relevant condition descriptions may be applicable to the Primary and Expansion components listed in Tables 5-2 and 5-3.

The examination acceptance criteria for components requiring visual (VT-3) examination is thus the absence of any of the relevant condition(s) specified in Table 5-5.

The disposition can include a supplementary examination to further characterize the relevant condition, an engineering evaluation to show that the component is capable of continued operation with a known relevant condition, or repair/replacement to remediate the relevant condition.

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#### **4.1.2 Visual (VT-1) Examination**

Visual (VT-1) examination is defined in the ASME Code Section XI as an examination “conducted to detect discontinuities and imperfections on the surface of components, including such conditions as cracks, wear, corrosion, or erosion.” The acceptance criterion for any visual (VT-1) examinations is the absence of any relevant conditions defined by the ASME Code, as supplemented by more specific plant inservice inspection requirements.

#### **4.1.3 Enhanced Visual (EVT-1) Examination**

Enhanced visual (EVT-1) examination has the same requirements as the ASME Code Section XI visual (VT-1) examination, with additional requirements given in the Inspection Standard, MRP-228. These enhancements are intended to improve the detection and characterization of discontinuities taking into account the remote visual aspect of reactor internals examinations. As a result, EVT-1 examinations are capable of detecting small surface breaking cracks and sizing surface crack length when used in conjunction with sizing aids (e.g. landmarks, ruler, and tape measure). EVT-1 examination is the appropriate NDE method for detection of cracking in plates or their welded joints. Thus the relevant condition applied for EVT-1 examination is the same as for cracking in Section XI which is crack-like surface breaking indications.

Therefore, until such time as engineering studies provide a basis by which a quantitative amount of degradation can be shown acceptable for the specific component, any observed relevant condition must be dispositioned. In the interim, the examination acceptance criterion is the absence of any detectable surface breaking indication.

#### **4.1.4 Surface Examination**

Surface ET (eddy current testing) examination is specified as an alternative or as a supplement to visual examinations. No specific acceptance criteria for surface (ET) examination of PWR internals locations are provided in the ASME Code Section XI. Since surface ET is employed as a signal-based examination, a technical justification per the Inspection Standard, MRP-228 provides the basis for detection and length sizing of surface-breaking or near-surface cracks. The signal-based relevant indication for surface (ET) is thus the same as the relevant condition for enhanced visual (EVT-1) examination. The acceptance criteria for enhanced visual (EVT-1) examinations in 4.1.3 (and accompanying entries in Table 5-5) are therefore applied when this method is used as an alternative or supplement to visual examination.

#### **4.1.5 Volumetric Examination**

The intent of volumetric examinations specified for bolts and pins is to detect planar defects. No flaw sizing measurements are recorded or assumed in the acceptance or rejection of individual

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bolts or pins. Individual bolts or pins are accepted based on the absence of relevant indications established as part of the examination technical justification. When a relevant indication is detected in the cross-sectional area of the bolt or pin, it is assumed to be non-functional and the indication is recorded. A bolt or pin that passes the criterion of the examination is considered functional.

Because of this pass/fail acceptance of individual bolts or pins, the examination acceptance criterion for volumetric (UT) examination of bolts and pins is based on a reliable detection of indications as established by the individual technical justification for the proposed examination. This is in keeping with current industry practice. For example, planar flaws on the order of 30% of the cross-sectional area have been determined reliably detectable in previous bolt NDE technical justifications for baffle-former bolting.

Bolted and pinned assemblies are evaluated for acceptance based on a plant specific evaluation.

#### **4.2 Physical Measurements Examination Acceptance Criteria**

Continued functionality can be confirmed by physical measurements where, for example, loss of material caused by wear, loss of pre-load of clamping force caused by various degradation mechanisms, or distortion/deflection caused by void swelling may occur. For Westinghouse designs, tolerances are available on a design or plant-specific basis. Specific acceptance criteria will be developed as required, and thus are not provided generically in this plan.

#### **4.3 Expansion Criteria**

The criterion for expanding the scope of examination from the Primary components to their linked Expansion components is contained in Table 5-5 for IPEC.

#### **4.4 Implementation Requirements**

4.2.1 Consistent with the requirements of NEI 03-08, if the guidance contained in Tables 5-2, 5-3, 5-4, and 5-5 cannot, need not, or will not be implemented as written, a technical justification must be prepared that clearly states what requirement cannot, need not, or will not be met and why; what alternative action is being taken to satisfy the objective or intent of the guidance; and why the alternative action is acceptable. Since the Expansion components are also "needed" requirements, the technical justification for not fully implementing a Primary component examination or not implementing it in a manner consistent with its intent, would be expected to include disposition of the associated Expansion components.

When submittal of a deviation from work products or elements is required, the justification shall be reviewed and approved in accordance with the applicable plant procedures with the additional responsibility for deviation from a "Needed" element that



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an internal independent review is performed and that concurrence is obtained from the responsible utility executive.

- 4.2.2 Examinations contained in this inspection plan shall be conducted in accordance with MRP-228.
- 4.2.3 Examination results that do not meet the examination acceptance criteria shall be recorded and entered in the IPEC corrective action program and dispositioned.
- 4.2.4 If an engineering evaluation is used to disposition an examination result that does not meet the examination acceptance criteria, this engineering evaluation shall be conducted in accordance with a NRC-approved evaluation methodology.
- 4.2.5 A summary report of all inspections and monitoring, items requiring evaluation, and new repairs shall be provided to the MRP Program Manager within 120 days of the completion of an outage during which PWR internals within the scope of MRP-227-A are examined.

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# 5 TABLES

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Table 5-1	Indian Point 2 & 3 Component Cross Reference
Table 5-2	Primary Components at IPEC Units 2 and 3
Table 5-3	Expansion Components at IPEC Units 2 and 3
Table 5-4	Existing Program Components at IPEC Units 2 and 3
Table 5-5	Examination Acceptance and Expansion Criteria at IPEC Units 2 and 3
Table 5-6	Reactor Vessel Component ISI Program Inspection Plan for IPEC Units 2 and 3
Table 5-7	List of IPEC Reactor Vessel Interior Components and Materials Based on MRP-191 – Table 4-4
Table 5-8	IPEC Response to the NRC Revision 1 to the Final Safety Evaluation of MRP-227

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**Table 5-1  
Indian Point 2 & 3 Component Cross Reference**

<b>Item</b>	<b>Letter NL-10-063 Component</b>	<b>MRP-191 Table 4-4</b>	<b>MRP-227-A</b>
1	Core Baffle/Former Assembly – Bolts	Lower Internals Assembly – Baffle and Former Assembly  Baffle-Edge Bolts  Baffle-Former Bolts	Baffle-Former Assembly – Baffle-Edge Bolts (Tables 3-3, 4-3 and 5-3)  Baffle-Former Assembly – Baffle-Former Bolts (Tables 3-3, 4-3 and 5-3)
2	Core Baffle/Former Assembly – Plates	Lower Internals Assembly – Baffle and Former Assembly  Baffle Plates  Former Plates	Baffle-Former Assembly – Assembly (Tables 3-3, 4-3 and 5-3)
3	Core Barrel Assembly – Bolts and Screws	Lower Internals Assembly – Baffle and Former Assembly  Barrel-Former Bolts	Core Barrel Assembly – Barrel-Former Bolts (Tables 3-3 and 4-6)
4	Core Barrel Assembly – Axial Flexure Plates (Thermal Shield Flexures)	Lower Internals Assembly – Neutron Panels/Thermal Shield  Thermal Shield Flexures	Thermal Shield Assembly – Thermal Shield Flexures (Tables 3-3, 4-3 and 5-3)
5	Core Barrel Assembly – Flange	Lower Internals Assembly – Core Barrel  Core Barrel Flange	Core Barrel Assembly – Core Barrel Flange (Tables 3-3 and 4-9)

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**Table 5-1  
Indian Point 2 & 3 Component Cross Reference**

<b>Item</b>	<b>Letter NL-10-063 Component</b>	<b>MRP-191 Table 4-4</b>	<b>MRP-227-A</b>
6	Core Barrel Assembly – Ring Core Barrel Assembly – Shell Core Barrel Assembly – Thermal Shield	Lower Internals Assembly – Core Barrel Upper Core Barrel Lower Core Barrel Lower Internals Assembly – Neutron panels/thermal shield Thermal shield	Core Barrel Assembly – Upper and Lower Core Barrel Cylinder Girth Welds (Tables 3-3, 4-3 and 2 places in 5-3) Core Barrel Assembly – Upper and Lower Core Barrel Cylinder Axial Welds (Tables 3-3, 4-6 and 2 places in 5-3)
7	Core Barrel Assembly – Lower Core Barrel Flange Weld Core Barrel Assembly – Upper Core Barrel Flange Weld	Lower Internals Assembly – Core Barrel Core Barrel Flange	Core Barrel Assembly – Lower Core Barrel Flange Weld (Tables 3-3, 4-3 and 5-3) Core Barrel Assembly – Upper Core Barrel Flange Weld (Tables 3-3, 4-3 and 5-3)
8	Core Barrel Assembly – Outlet Nozzles	Lower Internals Assembly – Core Barrel Core Barrel Outlet Nozzles	Core Barrel Assembly – Core Barrel Outlet Nozzle Welds (Tables 3-3 and 4-6)
9	Lower Internals Assembly – Clevis Insert Bolt	Interfacing Components – Interfacing Components Clevis Insert Bolts	Alignment and Interfacing Components – Clevis Insert Bolts (Tables 3-3 and 4-9)

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**Table 5-1  
 Indian Point 2 & 3 Component Cross Reference**

<b>Item</b>	<b>Letter NL-10-063 Component</b>	<b>MRP-191 Table 4-4</b>	<b>MRP-227-A</b>
10	Lower Internals Assembly – Clevis Insert	Interfacing Components – Interfacing Components  Clevis Inserts	No additional measures
11	Lower Internals Assembly – Intermediate Diffuser Plate	Lower Internals Assembly – Diffuser Plate  Diffuser Plate	No additional measures
12	Lower Internals Assembly – Fuel Alignment Pin	Lower Internals Assembly – Lower Core Plate and Fuel Alignment Pins  Fuel Alignment Pins	No additional measures
13	Lower Internals Assembly – Lower Core Plate	Lower Internals Assembly – Lower Core Plate and Fuel Alignment Pins  Lower Core Plate	Lower Internals Assembly – Lower Core Plate (Tables 3-3, and 2 places in 4-9)

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**Table 5-1  
Indian Point 2 & 3 Component Cross Reference**

Item	Letter NL-10-063 Component	MRP-191 Table 4-4	MRP-227-A
14	Lower Internals Assembly – <ul style="list-style-type: none"> <li>• Lower Core Support Castings</li> <li>• Column Cap</li> <li>• Lower Core Support Column Bodies</li> </ul>	Lower Internals Assembly – Lower Support Casting or Forging  Lower Support Casting  Lower Internals Assembly – Lower Support Column Assembly  Lower Support Column Nuts  Lower Support Column Bodies	Lower Internals Assembly – Lower Support Casting (Tables 3-3, and 4-6)    No additional measures  Lower Support Assembly – Lower Support Column Bodies (Cast) (Tables 3-3 and 4-6)
15	Lower Internals Assembly – Lower Core Support Plate Column Bolt	Lower Internals Assembly – Lower Support Column Assembly  Lower Support Column Bolts	Lower Support Assembly – Lower Support Column Bolts (Tables 3-3 and 4-6)
16	Lower Internals Assembly – Lower Core Support Plate Column Sleeves	Lower Internals Assembly – Lower Support Column Assembly  Lower Support Column Sleeves	No additional measures

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**Table 5-1  
Indian Point 2 & 3 Component Cross Reference**

<b>Item</b>	<b>Letter NL-10-063 Component</b>	<b>MRP-191 Table 4-4</b>	<b>MRP-227-A</b>
17	Lower Internals Assembly – Radial Key	Lower Internals Assembly – Radial Support Keys  Radial Support Keys	No additional measures
18	Lower Internals Assembly – Secondary Core Support	Lower Internals Assembly – Secondary Core Support (SCS) Assembly  SCS Base Plate	No additional measures
19	RCCA Guide Tube Assembly – Bolt	Upper Internals Assembly – Control Rod Guide Tube Assemblies and Flow Downcomers  Bolts	No additional measures
20	RCCA Guide Tube Assembly – Guide Tube (including Lower Flange Welds)	Upper Internals Assembly – Control Rod Guide Tube Assemblies and Flow Downcomers  Flanges – lower	Control Rod Guide Tube Assembly – Lower Flange Welds (Tables 3-3, 4-3 and 5-3)
21	RCCA Guide Tube Assembly – Guide Plates	Upper Internals Assembly – Control Rod Guide Tube Assemblies and Flow Downcomers  Guide Plates/Cards	Control Rod Guide Tube Assembly – Guide Plates (Cards) (Tables 3-3, 4-3 and 5-3)

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**Table 5-1  
Indian Point 2 & 3 Component Cross Reference**

<b>Item</b>	<b>Letter NL-10-063 Component</b>	<b>MRP-191 Table 4-4</b>	<b>MRP-227-A</b>
22	RCCA Guide Tube Assembly – Support Pin	Upper Internals Assembly – Control Rod Guide Tube Assemblies and Flow Downcomers.  Guide Tube Support Pins	No additional measures
23	Core Plate Alignment Pin	Interfacing Components – Interfacing Components  Upper Core Plate Alignment Pins	Alignment and Interfacing Components – Upper Core Plate Alignment Pins (Tables 3-3 and 4-9)
24	Head/Vessel Alignment Pin	Interfacing Components – Interfacing Components  Head and Vessel Alignment Pins	No additional measures
25	Hold-down Spring	Interfacing Components – Interfacing Components  Internals Hold Down Spring	Alignment and Interfacing Components – Internals Hold Down Spring (Tables 3-3, 4-3 and 5-3)
26	Mixing Devices - Support Column Orifice Base - Support Column Mixer	Upper Internals Assembly – Mixing Devices  Mixing devices	No additional measures



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**Table 5-1  
Indian Point 2 & 3 Component Cross Reference**

<b>Item</b>	<b>Letter NL-10-063 Component</b>	<b>MRP-191 Table 4-4</b>	<b>MRP-227-A</b>
27	Support Column	Upper Internals Assembly – Upper Support Column Assemblies  Column Bodies	No additional measures
28	Upper Core Plate, Fuel Alignment Pin	Upper Internals Assembly – Upper Core Plate and Fuel Alignment Pins  Fuel Alignment Pins	No additional measures
29	Upper Support Plate, Support Assembly (Including Ring)	Upper Internals Assembly – Upper Support Plate Assembly  Upper Support Plate Upper Support Ring or Skirt	No additional measures for the upper support plate  Upper Internals Assembly – Upper Support Ring or Skirt (Tables 3-3 and 4-9)
30	Upper Support Column Bolt	Upper Internals Assembly – Upper Support Column Assemblies  Bolts	No additional measures
31	Bottom Mounted Instrumentation Column	Lower Internals Assembly – Bottom-Mounted Instrumentation (BMI) Column Assemblies  BMI Column Bodies	Bottom Mounted Instrumentation System – Bottom Mounted Instrumentation (BMI) Column Bodies (Tables 3-3 and 4-6)

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**Table 5-1  
 Indian Point 2 & 3 Component Cross Reference**

<b>Item</b>	<b>Letter NL-10-063 Component</b>	<b>MRP-191 Table 4-4</b>	<b>MRP-227-A</b>
32	Flux Thimble Guide Tube	Lower Internals Assembly – Flux Thimbles (Tubes)  Flux Thimbles (Tubes)	Bottom Mounted Instrumentation System – Flux Thimble Tubes (Tables 3-3 and 4-9)
33	Thermocouple Conduit	Upper Internals Assembly – Upper Instrumentation Conduit and Support  Conduits	No additional measures

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**Table 5-2  
 Primary Components at IPEC Units 2 and 3**

Item	Applicability	Effect (Mechanism)	Expansion Link	Examination Method/Frequency	Examination Coverage
<b>Control Rod Guide Tube Assembly</b> Guide plates (cards)	IPEC Units 2 and 3	Loss of Material (Wear)	None	Visual (VT-3) examination no later than 2 refueling outages from the beginning of the license renewal period. Subsequent examinations are required on a ten-year interval.	20% examination of the number of CRGT assemblies, with all guide cards within each selected CRGT assembly examined.  See Figure 2-2
<b>Control Rod Guide Tube Assembly</b> Lower flange welds	IPEC Units 2 and 3	Cracking (SCC, Fatigue) Aging Management (IE and TE)	Bottom-mounted instrumentation (BMI) column bodies, Lower support column bodies (cast) Upper core plate Lower support casting	Enhanced visual (EVT-1) examination to determine the presence of crack-like surface flaws in flange welds no later than 2 refueling outages from the beginning of the license renewal period and subsequent examination on a ten-year interval.	100% of outer (accessible) CRGT lower flange weld surfaces and adjacent base metal on the individual periphery CRGT assemblies. A minimum of 75% of the total identified sample population must be examined.  See Figure 2-3

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**Table 5-2  
Primary Components at IPEC Units 2 and 3**

<b>Item</b>	<b>Applicability</b>	<b>Effect (Mechanism)</b>	<b>Expansion Link</b>	<b>Examination Method/Frequency</b>	<b>Examination Coverage</b>
<b>Core Barrel Assembly</b> Upper core barrel flange weld	IPEC Units 2 and 3	Cracking (SCC)	Core barrel outlet nozzle welds	Periodic enhanced visual (EVT-1) examination, no later than 2 refueling outages from the beginning of the license renewal period and subsequent examination on a ten-year interval.	100% of one side of the accessible surfaces of the selected weld and adjacent base metal. A minimum of 75% of the total weld length (examined + unexamined), including coverage consistent with the Expansion criteria in Table 5-5, must be examined from either the inner or outer diameter for inspection credit.  See Figure 2-4
<b>Core Barrel Assembly</b> Upper and lower core barrel cylinder girth welds	IPEC Units 2 and 3	Cracking (SCC, IASCC, Fatigue)	Upper and lower core barrel cylinder axial welds	Periodic enhanced visual (EVT-1) examination, no later than 2 refueling outages from the beginning of the license renewal period and subsequent examination on a ten-year interval.	100% of one side of the accessible surfaces of the selected weld and adjacent base metal. A minimum of 75% of the total weld length (examined + unexamined), including coverage consistent with the Expansion criteria in Table 5-5, must be examined from either the inner or outer diameter for inspection credit.  See Figure 2-4

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**Table 5-2  
 Primary Components at IPEC Units 2 and 3**

Item	Applicability	Effect (Mechanism)	Expansion Link	Examination Method/Frequency	Examination Coverage
<p><b>Core Barrel Assembly</b>            Lower core barrel flange weld</p> <p>(At IPEC this weld is the lower core barrel to lower support casting weld. IPEC does not have a lower core barrel flange)</p>	<p>IPEC Units 2 and 3</p>	<p>Cracking (SCC, Fatigue)</p>	<p>None</p>	<p>Periodic enhanced visual (EVT-1) examination, no later than 2 refueling outages from the beginning of the license renewal period and subsequent examination on a ten-year interval.</p>	<p>100% of one side of the accessible surfaces of the selected weld and adjacent base metal. A minimum of 75% of the total weld length (examined + unexamined), including coverage consistent with the Expansion criteria in Table 5-5, must be examined from either the inner or outer diameter for inspection credit.</p> <p>See Figure 2-4 (Core Barrel to Support Plate Weld)</p>

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**Table 5-2  
 Primary Components at IPEC Units 2 and 3**

Item	Applicability	Effect (Mechanism)	Expansion Link	Examination Method/Frequency	Examination Coverage
<b>Baffle-Former Assembly</b> Baffle-edge bolts	IPEC Units 2 and 3	Cracking (IASCC, Fatigue) that results in <ul style="list-style-type: none"> <li>• Lost or broken locking devices</li> <li>• Failed or missing bolts</li> <li>• Protrusion of bolt heads</li> </ul> Aging Management (IE and ISR) Void swelling effects on this component is managed through management of void swelling on the entire baffle-former assembly.	None	Visual (VT-3) examination, with baseline examination between 20 and 40 EFPY and subsequent examinations on a ten-year interval.	Bolts and locking devices on high fluence seams. 100% of components accessible from core side. A minimum of 75% of the total population (examined + unexamined), including coverage consistent with the Expansion criteria in Table 5-5, must be examined for inspection credit.  See Figure 2-5

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**Table 5-2  
 Primary Components at IPEC Units 2 and 3**

Item	Applicability	Effect (Mechanism)	Expansion Link	Examination Method/Frequency	Examination Coverage
<b>Baffle-Former Assembly</b> Baffle-former bolts	IPEC Units 2 and 3	Cracking (IASCC, Fatigue) Aging management (IE and ISR) Void swelling effects on this component is managed through management of void swelling on the entire baffle- former assembly.	Lower support column bolts, Barrel-former bolts	Baseline volumetric (UT) examination between 25 and 35 EFPY, with subsequent examination on a ten-year interval.  <u>See additional IPEC specific            examination requirements in            Section 6.2.</u>	100% of accessible bolts. A minimum of 75% of the total population (examined + unexamined), including coverage consistent with the Expansion criteria in Table 5-5, must be examined for inspection credit. Heads accessible from the core side. UT accessibility may be affected by complexity of head and locking device designs.  See Figures 2-5 and 2-6.

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**Table 5-2  
Primary Components at IPEC Units 2 and 3**

Item	Applicability	Effect (Mechanism)	Expansion Link	Examination Method/Frequency	Examination Coverage
<b>Baffle-Former Assembly</b> Assembly (Includes: Baffle plates, baffle edge bolts and indirect effects of void swelling in former plates)	IPEC Units 2 and 3	Distortion (Void Swelling), or Cracking (IASCC) that results in <ul style="list-style-type: none"> <li>• Abnormal interaction with fuel assemblies</li> <li>• Gaps along high fluence baffle joint</li> <li>• Vertical displacement of baffle plates near high fluence joint</li> <li>• Broken or damaged edge bolt locking systems along high fluence baffle joint</li> </ul>	None	Visual (VT-3) examination to check for evidence of distortion, with baseline examination between 20 and 40 EFPY and subsequent examinations on a ten-year interval.	Core side surface as indicated.  See Figures 2-6, 2-7, 2-8 and 2-9.
<b>Alignment and Interfacing Components</b> Internals hold down spring	IPEC Units 2 and 3	Distortion (Loss of Load)  Note: This mechanism was not strictly identified in the original list of age-related degradation mechanisms.	None	Direct measurement of spring height within three cycles of the beginning of the license renewal period. If the first set of measurements is not sufficient to determine life, spring height measurements must be taken during the next two outages, in order to extrapolate the expected spring height to 60 years.	Measurements should be taken at several points around the circumference of the spring, with a statistically adequate number of measurements at each point to minimize uncertainty.  See Figure 2-10



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**Table 5-2  
 Primary Components at IPEC Units 2 and 3**

Item	Applicability	Effect (Mechanism)	Expansion Link	Examination Method/Frequency	Examination Coverage
<b>Thermal Shield Assembly</b> Thermal shield flexures  >	IPEC Units 2 and 3	Cracking (Fatigue) or Loss of Materials (Wear) that results in thermal shield flexures excessive wear, fracture or complete separation	None	Visual (VT-3) no later than 2 refueling outages from the beginning of the license renewal period. Subsequent examinations on a ten year interval.	100% of thermal shield flexures  See Figures 2-11 and 2-16

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**Table 5-3  
 Expansion Components at IPEC Units 2 and 3**

Item	Applicability	Effect (Mechanism)	Primary Link	Examination Method	Examination Coverage
<b>Upper Internals Assembly</b> Upper core plate	IPEC Units 2 and 3	Cracking (Fatigue, Wear)	Control rod guide tube (CRGT) lower flange weld	Enhanced visual (EVT-1) examination.  Re-inspection every 10 years following initial inspection.	100% of accessible surfaces. A minimum of 75% coverage of the entire examination area or volume, or a minimum sample size of 75% of the total population of like components of the examination is required (including both the accessible and inaccessible portions).  See Figure 2-1
<b>Lower Internals Assembly</b> Lower support casting	IPEC Units 2 and 3	Cracking Aging Management (TE in Casting)	Control rod guide tube (CRGT) lower flange weld	Enhanced visual (EVT-1) examination.  Re-inspection every 10 years following initial inspection.	100% of accessible surfaces. A minimum of 75% coverage of the entire examination area or volume, or a minimum sample size of 75% of the total population of like components of the examination is required (including both the accessible and inaccessible portions).  See Figure 2-1 (Core Support)

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**Table 5-3  
 Expansion Components at IPEC Units 2 and 3**

Item	Applicability	Effect (Mechanism)	Primary Link	Examination Method	Examination Coverage
<b>Core Barrel Assembly</b> Barrel-former bolts	IPEC Units 2 and 3	Cracking (IASCC, Fatigue)  Aging Management (IE, Void Swelling and ISR)	Baffle-former bolts	Volumetric (UT) examination.  Re-inspection every 10 years following initial inspection.	100% of accessible bolts. Accessibility may be limited by presence of thermal shields. A minimum of 75% coverage of the entire examination area or volume, or a minimum sample size of 75% of the total population of like components of the examination is required (including both the accessible and inaccessible portions).  See Figure 2-5.

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**Table 5-3  
 Expansion Components at IPEC Units 2 and 3**

Item	Applicability	Effect (Mechanism)	Primary Link	Examination Method	Examination Coverage
<b>Lower Support Assembly</b> Lower support column bolts	IPEC Units 2 and 3	Cracking (IASCC, Fatigue)  Aging Management (IE, and ISR)	Baffle-former bolts	Volumetric (UT) examination.  Re-inspection every 10 years following initial inspection.	100% of accessible bolts or as supported by plant-specific justification. A minimum of 75% coverage of the entire examination area or volume, or a minimum sample size of 75% of the total population of like components of the examination is required (including both the accessible and inaccessible portions)  See Figures 2-12 and 2-13
<b>Core Barrel Assembly</b> Core barrel outlet nozzle welds	IPEC Units 2 and 3	Cracking (SCC, Fatigue)  Aging Management (IE of lower sections)	Upper core barrel flange weld	Enhanced visual (EVT-1) examination.  Re-inspection every 10 years following initial inspection.	100% of one side of the accessible surfaces of the selected weld and adjacent base metal. A minimum of 75% coverage of the entire examination area or volume, or a minimum sample size of 75% of the total population of like components of the examination is required (including both the accessible and inaccessible portions)  See Figure 2-4

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**Table 5-3  
 Expansion Components at IPEC Units 2 and 3**

Item	Applicability	Effect (Mechanism)	Primary Link	Examination Method	Examination Coverage
<b>Core Barrel Assembly</b> Upper and lower core barrel cylinder axial welds	IPEC Units 2 and 3	Cracking (SCC, IASCC)  Aging Management (IE)	Upper and lower core barrel cylinder girth welds	Enhanced visual (EVT-1) examination.  Re-inspection every 10 years following initial inspection.	100% of one side of the accessible surfaces of the selected weld and adjacent base metal. A minimum of 75% coverage of the entire examination area or volume, or a minimum sample size of 75% of the total population of like components of the examination is required (including both the accessible and inaccessible portions)  See Figure 2-4
<b>Lower Support Assembly</b> Lower support column bodies (non cast)	IPEC lower support column bodies are cast. They are captured in the next Item of this table.				

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**Table 5-3  
 Expansion Components at IPEC Units 2 and 3**

Item	Applicability	Effect (Mechanism)	Primary Link	Examination Method	Examination Coverage
<b>Lower Support Assembly</b> Lower support column bodies (cast)	IPEC Units 2 and 3	Cracking (IASCC) including the detection of fractured support columns  Aging Management (IE)	Control rod guide tube (CRGT) lower flanges	Enhanced visual (EVT-1) examination.  Re-inspection every 10 years following initial inspection.	100% of accessible support columns. A minimum of 75% coverage of the entire examination area or volume, or a minimum sample size of 75% of the total population of like components of the examination is required (including both the accessible and inaccessible portions)  See Figure 2-14
<b>Bottom Mounted Instrumentation System</b> Bottom-mounted instrumentation (BMI) column bodies	IPEC Units 2 and 3	Cracking (Fatigue) including the detection of completely fractured column bodies  Aging Management (IE)	Control rod guide tube (CRGT) lower flanges	Visual (VT-3) examination of BMI column bodies as indicated by difficulty of insertion/withdrawal of flux thimbles.  Re-inspection every 10 years following initial inspection.  Flux thimble insertion/withdrawal to be monitored at each inspection interval.	100% of BMI column bodies for which difficulty is detected during flux thimble insertion/withdrawal.  See Figure 2-15

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**Table 5-4  
Existing Program Components at IPEC Units 2 and 3**

<b>Item</b>	<b>Applicability</b>	<b>Effect (Mechanism)</b>	<b>Reference</b>	<b>Examination Method</b>	<b>Examination Coverage</b>
<b>Core Barrel Assembly</b> Core barrel flange	IPEC Units 2 and 3	Loss of material (Wear)	ASME Code Section XI	Visual (VT-3) examination to determine general condition for excessive wear.	All accessible surfaces at specified frequency.
<b>Upper Internals Assembly</b> Upper support ring or skirt  IPEC has a tophat design, therefore there is no support ring or skirt, however the vertical sections of the tophat will be inspected	IPEC Units 2 and 3	Cracking (SCC, Fatigue)	ASME Code Section XI	Visual (VT-3) examination.	All accessible surfaces at specified frequency.
<b>Lower Internals Assembly</b> Lower core plate	IPEC Units 2 and 3	Cracking (IASCC, Fatigue) Aging Management (IE)	ASME Code Section XI	Visual (VT-3) examination of the lower core plates to detect evidence of distortion and/or loss of bolt integrity.	All accessible surfaces at specified frequency.
<b>Lower Internals Assembly</b> Lower core plate	IPEC Units 2 and 3	Loss of material (Wear)	ASME Code Section XI	Visual (VT-3) examination.	All accessible surfaces at specified frequency.
<b>Bottom Mounted Instrumentation System</b> Flux thimble tubes	IPEC Units 2 and 3	Loss of material (Wear)	NUREG-1801 Rev. 1	Surface (ET) examination.	Eddy current surface examination as defined in plant response to IEB 88-09
<b>Alignment and Interfacing Components</b> Clevis insert bolts	IPEC Units 2 and 3	Loss of material (Wear)	ASME Code Section XI	Visual (VT-3) examination.	All accessible surfaces at specified frequency.
<b>Alignment and Interfacing Components</b> Upper core plate alignment pins	IPEC Units 2 and 3	Loss of material (Wear)	ASME Code Section XI	Visual (VT-3) examination.	All accessible surfaces at specified frequency.

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**Table 5-5  
Examination Acceptance and Expansion Criteria at IPEC Units 2 and 3**

<b>Item</b>	<b>Applicability</b>	<b>Examination Acceptance Criteria (Note 1)</b>	<b>Expansion Link(s)</b>	<b>Expansion Criteria</b>	<b>Additional Examination Acceptance Criteria</b>
<b>Control Rod Guide Tube Assembly</b>  Guide plates (cards)	IPEC Units 2 and 3	Visual (VT-3) examination.  The specific relevant condition is wear that could lead to loss of control rod alignment and impede control assembly insertion.	None	N/A	N/A
<b>Control Rod Guide Tube Assembly</b>  Lower flange welds	IPEC Units 2 and 3	Enhanced visual (EVT-1) examination.  The specific relevant condition is a detectable crack-like surface indication.	a. Bottom-mounted instrumentation (BMI) column bodies  b. Lower support column bodies (cast), upper core plate and lower support casting	a. Confirmation of surface-breaking indications in two or more CRGT lower flange welds, combined with flux thimble insertion/withdrawal difficulty, shall require visual (VT-3) examination of BMI column bodies by the completion of the next refueling outage.  b. Confirmation of surface-breaking indications in two or more CRGT lower flange welds shall require EVT-1 examination of cast lower support column bodies, upper core plate and lower support casting within three fuel cycles following the initial observation.	a. For BMI column bodies, the specific relevant condition for the VT-3 examination is completely fractured column bodies.  b. For cast lower support column bodies, upper core plate and lower support casting, the specific relevant condition is a detectable crack-like surface indication.



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**Table 5-5  
 Examination Acceptance and Expansion Criteria at IPEC Units 2 and 3**

Item	Applicability	Examination Acceptance Criteria (Note 1)	Expansion Link(s)	Expansion Criteria	Additional Examination Acceptance Criteria
<b>Core Barrel Assembly</b> Upper core barrel flange weld	IPEC Units 2 and 3	Periodic enhanced visual (EVT-1) examination.  The specific relevant condition is a detectable crack-like surface indication.	a. Core barrel outlet nozzle welds  b. Lower support column bodies (non cast)  IPEC lower support column bodies are cast	a. The confirmed detection and sizing of a surface-breaking indication with a length greater than two inches in the upper core barrel flange weld shall require that the EVT-1 examination be expanded to include the core barrel outlet nozzle welds by the completion of the next refueling outage.  b. N/A	a. The specific relevant condition for the expansion core barrel outlet nozzle weld examination is a detectable crack-like surface indication.  b. N/A
<b>Core Barrel Assembly</b> Lower core barrel flange weld (At IPEC this weld is the lower core barrel to lower support casting weld. IPEC does not have a lower core barrel flange.)	IPEC Units 2 and 3	Periodic enhanced visual (EVT-1) examination.  The specific relevant condition is a detectable crack-like surface indication.	None	None	None

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**Table 5-5  
 Examination Acceptance and Expansion Criteria at IPEC Units 2 and 3**

<b>Item</b>	<b>Applicability</b>	<b>Examination Acceptance Criteria (Note 1)</b>	<b>Expansion Link(s)</b>	<b>Expansion Criteria</b>	<b>Additional Examination Acceptance Criteria</b>
<b>Core Barrel Assembly</b> Upper core barrel cylinder girth welds	IPEC Units 2 and 3	Periodic enhanced visual (EVT-1) examination.  The specific relevant condition is a detectable crack-like surface indication.	Upper core barrel cylinder axial welds	The confirmed detection and sizing of a surface-breaking indication with a length greater than two inches in the upper core barrel cylinder girth welds shall require that the EVT-1 examination be expanded to include the upper core barrel cylinder axial welds by the completion of the next refueling outage.	The specific relevant condition for the expansion upper core barrel cylinder axial weld examination is a detectable crack-like surface indication.
<b>Core Barrel Assembly</b> Lower core barrel cylinder girth welds	IPEC Units 2 and 3	Periodic enhanced visual (EVT-1) examination.  The specific relevant condition is a detectable crack-like surface indication.	Lower core barrel cylinder axial welds	The confirmed detection and sizing of a surface-breaking indication with a length greater than two inches in the lower core barrel cylinder girth welds shall require that the EVT-1 examination be expanded to include the lower core barrel cylinder axial welds by the completion of the next refueling outage.	The specific relevant condition for the expansion lower core barrel cylinder axial weld examination is a detectable crack-like surface indication.
<b>Baffle-Former Assembly</b> Baffle-edge bolts	IPEC Units 2 and 3	Visual (VT-3) examination.  The specific relevant conditions are missing or broken locking devices, failed or missing bolts, and protrusion of bolt heads.	None	N/A	N/A

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**Table 5-5  
 Examination Acceptance and Expansion Criteria at IPEC Units 2 and 3**

Item	Applicability	Examination Acceptance Criteria (Note 1)	Expansion Link(s)	Expansion Criteria	Additional Examination Acceptance Criteria
<b>Baffle-Former Assembly</b> Baffle-former bolts	IPEC Units 2 and 3	Volumetric (UT) examination.  The examination acceptance criteria for the UT of the baffle-former bolts shall be established as part of the examination technical justification.	a. Lower support column bolts  b. Barrel-former bolts	a. Confirmation that more than 5% of the baffle-former bolts actually examined on the four baffle plates at the largest distance from the core (presumed to be the lowest dose locations) contain unacceptable indications shall require UT examination of the lower support column bolts within the next three fuel cycles.  b. Confirmation that more than 5% of the lower support column bolts actually examined contain unacceptable indications shall require UT examination of the barrel-former bolts.	a and b. The examination acceptance criteria for the UT of the lower support column bolts and the barrel-former bolts shall be established as part of the examination technical justification.

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**Table 5-5  
 Examination Acceptance and Expansion Criteria at IPEC Units 2 and 3**

Item	Applicability	Examination Acceptance Criteria (Note 1)	Expansion Link(s)	Expansion Criteria	Additional Examination Acceptance Criteria
<b>Baffle-Former Assembly</b> Assembly	IPEC Units 2 and 3	Visual (VT-3) examination.  The specific relevant conditions are evidence of abnormal interaction with fuel assemblies, gaps along high fluence shroud plate joints, vertical displacement of shroud plates near high fluence joints, and broken or damaged edge bolt locking systems along high fluence baffle plate joints.	None	N/A	N/A
<b>Alignment and Interfacing Components</b> Internals hold down spring	IPEC Units 2 and 3	Direct physical measurement of spring height.  The examination acceptance criterion for this measurement is that the remaining compressible height of the spring shall provide hold-down forces within the plant-specific design tolerance.	None	N/A	N/A

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**Table 5-5  
 Examination Acceptance and Expansion Criteria at IPEC Units 2 and 3**

Item	Applicability	Examination Acceptance Criteria (Note 1)	Expansion Link(s)	Expansion Criteria	Additional Examination Acceptance Criteria
<b>Thermal Shield Assembly</b> Thermal shield flexures	IPEC Units 2 and 3	Visual (VT-3) examination.  The specific relevant conditions for thermal shield flexures are excessive wear, fracture, or complete separation.	None	N/A	N/A

Notes:

1. The examination acceptance criterion for visual examination is the absence of the specified relevant condition

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**Table 5-6  
 Reactor Vessel Component ISI Program Inspection Plan for IPEC Units 2 and 3**

<b>Component</b>	<b>Code Category</b>	<b>Examination Method</b>	<b>Extent of Exam</b>
<b>Lower Internals - Exterior</b> Core barrel surface	B-N-3	VT-3	Components and areas as accessible
<b>Lower Internals - Exterior</b> Thermal Shield	B-N-3	VT-3	Components and areas as accessible
<b>Lower Internals - Exterior</b> Irradiation specimen tubes and guides	B-N-3	VT-3	Components and areas as accessible
<b>Lower Internals - Exterior</b> Flexures	B-N-3	VT-3	Components and areas as accessible
<b>Lower Internals - Exterior</b> Fasteners and locking devices	B-N-3	VT-3	Components and areas as accessible
<b>Lower Internals - Exterior</b> Outlet nozzles at 22 deg, 158 deg, 202 deg, and 338 deg	B-N-3	VT-3	Components and areas as accessible
<b>Lower Internals – Exterior Bottom</b> Lower core support plate	B-N-3	VT-3	Components and areas as accessible
<b>Lower Internals – Exterior Bottom</b> Flow distribution plate	B-N-3	VT-3	Components and areas as accessible
<b>Lower Internals – Exterior Bottom</b> Lower support casting	B-N-3	VT-3	Components and areas as accessible

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**Table 5-6  
 Reactor Vessel Component ISI Program Inspection Plan for IPEC Units 2 and 3**

<b>Component</b>	<b>Code Category</b>	<b>Examination Method</b>	<b>Extent of Exam</b>
<b>Lower Internals – Exterior Bottom</b> Core support column	B-N-3	VT-3	Components and areas as accessible
<b>Lower Internals – Exterior Bottom</b> Secondary core support	B-N-3	VT-3	Components and areas as accessible
<b>Lower Internals – Exterior Bottom</b> Instrumentation guides	B-N-3	VT-3	Components and areas as accessible
<b>Lower Internals – Exterior Bottom</b> Radial support keys	B-N-3	VT-3	Components and areas as accessible
<b>Lower Internals – Interior Bottom</b> Outlet nozzles at 22 deg, 158 deg, 202 deg, and 338 deg	B-N-3	VT-3	Components and areas as accessible
<b>Lower Internals – Interior Bottom</b> Core barrel alignment pin	B-N-3	VT-3	Components and areas as accessible
<b>Lower Internals – Interior Bottom</b> Lower core plate	B-N-3	VT-3	Components and areas as accessible
<b>Lower Internals – Interior Bottom</b> Fuel alignment pins	B-N-3	VT-3	Components and areas as accessible
<b>Upper Internals Assembly</b> Vertical sections of tophat	B-N-3	VT-3	Components and areas as accessible

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**Table 5-6  
 Reactor Vessel Component ISI Program Inspection Plan for IPEC Units 2 and 3**

<b>Component</b>	<b>Code Category</b>	<b>Examination Method</b>	<b>Extent of Exam</b>
<b>Core Barrel Assembly</b> Core barrel flange	B-N-3	VT-3	Components and areas as accessible
<b>Alignment and Interfacing Components</b> Clevis insert bolts	B-N-3	VT-3	Components and areas as accessible
<b>Alignment and Interfacing Components</b> Upper core plate alignment pins	B-N-3	VT-3	Components and areas as accessible



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**Table 5-7**  
**List of IPEC Reactor Vessel Interior Components and Materials Based on MRP-191 – Table 4-4**

UPPER INTERNALS ASSEMBLY			
Sub Assembly	Component	Material	Category from MRP-191 Table 7-2
Control rod guide tube assemblies and flow downcomers	Anti-rotation studs and nuts	Stainless steel	A
	Bolts	Stainless steel	A
	C-tubes	Stainless steel	C
	Enclosure pins	Stainless steel	A
	Upper guide tube enclosures	Stainless steel	A
	Flanges intermediate	Stainless steel	A
	Flanges lower	Stainless steel	A
	Flexureless inserts	Stainless steel	A
	Guide plates/cards	Stainless steel	C
	Guide tube support pins (split pins)	A X-750 (IP2 only)	C
	Guide tube support pins (split pins)	Stainless steel (IP3 only)	A
	Housing plates	Stainless steel	A
	Inserts	Stainless steel	A
	Lock bars	Stainless steel	A
	Sheaths	Stainless steel	C
	Support pin cover plate	Stainless steel	A
	Support pin cover plate cap screws	Stainless steel	A
	Support pin cover plate locking caps and tie straps	Stainless steel	A
	Support pin nuts	Alloy X-750 (IP2 only)	A
	Support pin nuts	Stainless steel (IP3 only)	A
Water flow slot ligaments	Stainless steel	A	
Mixing Devices	Mixing devices	CASS	A
Upper core plate and fuel alignment pins	Fuel alignment pins	Stainless steel	A
	Upper core plate	Stainless steel	A
Upper instrumentation conduit and supports	Bolting	Stainless steel	A
	Brackets,clamps,terminal blocks, and conduit straps	Stainless steel	A
	Conduit seal assembly-body, tubesheets	Stainless steel	A
	Conduit seal assembly-tubes	Stainless steel	A
	Conduits	Stainless steel	A
	Flange base	Stainless steel	A
	Locking caps	Stainless steel	A
	Support tubes	Stainless steel	A
Upper plenum	UHI flow column bases	CASS	A
	UHI flow columns	Stainless steel	A

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**Table 5-7**  
**List of IPEC Reactor Vessel Interior Components and Materials Based on MRP-191 – Table 4-4**

UPPER INTERNALS ASSEMBLY			
Sub Assembly	Component	Material	Category from MRP-191 Table 7-2
Upper support column assemblies	Adapters	Stainless steel	A
	Bolts	Stainless steel	A
	Column bases	CASS	A
	Column bodies	Stainless steel	A
	Extension tubes	Stainless steel	A
	Flanges	Stainless steel	A
	Lock keys	Stainless steel	A
	Nuts	Stainless steel	A
Upper support plate assembly	Bolts	Stainless steel	A
	Deep beam ribs	Stainless steel	A
	Deep beam stiffeners	Stainless steel	A
	Flange	Stainless steel	A
	Inverted top hat flange	Stainless steel	A
	Inverted top hat upper support plate	Stainless steel	A
	Lock keys	Stainless steel	A
	Ribs	Stainless steel	A
	Upper support plate	Stainless steel	A
Upper support ring or skirt	Stainless steel	B	
LOWER INTERNALS ASSEMBLY			
Sub Assembly	Component	Material	Category from MRP-191 Table 7-2
Baffle and former assembly	Baffle bolting locking bar	Stainless steel	A
	Baffle edge bolts	Stainless steel	C
	Baffle plates	Stainless steel	B
	Baffle former bolts	Stainless steel	C
	Barrel former bolts	Stainless steel	C
	Former plates	Stainless steel	B
Bottom mounted instrumentation (BMI) column assemblies	BMI column bodies	Stainless steel	B
	BMI column bolts	Stainless steel	A
	BMI column collars	Stainless steel	B
	BMI column cruciforms	CASS	B
	BMI column extension bars	Stainless steel	A
	BMI column extension tubes	Stainless steel	B
	BMI column lock caps	Stainless steel	A
BMI column nuts	Stainless steel	A	
Core barrel	Core barrel flange	Stainless steel	B
	Core barrel outlet nozzles	Stainless steel	B
	Upper core barrel	Stainless steel	C
	Lower core barrel	Stainless steel	C
Diffuser plate	Diffuser plate	Stainless steel	A

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**Table 5-7  
List of IPEC Reactor Vessel Interior Components and Materials Based on MRP-191 – Table 4-4**

LOWER INTERNALS ASSEMBLY			
Sub Assembly	Component	Material	Category from MRP-191 Table 7-2
Flux thimbles (tubes)	Flux thimble tube plugs - IPEC does not use tube plugs	Stainless steel	B
	Flux thimbles (tubes)	Stainless steel	C
Irradiation specimen guides	Irradiation specimen guide	Stainless steel	A
	Irradiation specimen guide bolts	Stainless steel	A
	Irradiation specimen lock caps	Stainless steel	A
	Specimen plugs	Stainless steel	A
Lower core plate (LCP) and fuel alignment pins	Fuel alignment pins	Stainless steel	A
	LCP fuel alignment pin bolts	Stainless steel	A
	LCP fuel alignment pin lock caps	Stainless steel	A
	Lower core plate	Stainless steel	C
Lower support column assemblies	Lower support column bodies	CASS	B
	Lower support column bolts	Stainless steel	B
	Lower support column nuts	Stainless steel	A
	Lower support column sleeves	Stainless steel	A
Lower support casting or forging	Lower support casting	CASS	A
Neutron panels/thermal shield	Thermal shield bolts	Stainless steel	A
	Thermal shield dowels	Stainless steel	A
	Thermal shield flexures	Stainless steel	B
	Thermal shield	Stainless steel	A
Radial support keys	Radial support key bolts	Stainless steel	A
	Radial support key lock keys	Stainless steel	A
	Radial support keys	Stainless steel	A
Secondary core support (SCS) assembly	SCS base plate	Stainless steel	A
	SCS bolts	Stainless steel	A
	SCS energy absorber	Stainless steel	A
	SCS guide posts	Stainless steel	A
	SCS housing	Stainless steel	A
	SCS lock keys	Stainless steel	A
Interfacing Components	Clevis insert bolts	A X-750	B
	Clevis insert lock keys	Stainless steel	A
	Clevis inserts	Alloy 600	A
	Head and vessel alignment pin bolts	Stainless steel	A
	Head and vessel alignment pin lock caps	Stainless steel	A
	Head and vessel alignment pins	Stainless steel	A
	Internals hold down spring	304 Stainless steel	B
	Upper core plate alignment pins	Stainless steel	B

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**Table 5-8  
IPEC Response to the NRC Revision 1 to the Final Safety Evaluation of MRP-227**

MRP-227-A SER Item	IPEC Response
SER Section 4.1.1, Topical Report Condition 1 Moving components to "Expansion" category from "No additional measures" category.	In accordance with SER Section 4.1.1, the upper core plate and the lower support casting have been added to the IPEC "Expansion" inspection category and are contained in Table 5-3. The components are linked to the "Primary" component CRGT lower flange weld. The examination method is consistent with the examinations performed on the CRGT lower flange weld.
SER Section 4.1.2, Topical Report Condition 2 Inspection of components subject to irradiation-assisted stress corrosion cracking	In accordance with SER Section 4.1.2, the upper and lower core barrel cylinder girth welds and lower core barrel to lower support casting weld have been added to the IPEC "Primary" inspection category and are contained in Table 5-2. The examination method is consistent with the MRP recommendations for these components, the examination coverage conforms to the criteria described in Section 3.3.1 of the NRC SE, and the re-examination frequency is on a 10-year interval consistent with other "Primary" inspection category components. The inspection shall be expanded to axial welds (expansion component) in the event that degradation is observed in the girth welds.
SER Section 4.1.3, Topical Report Condition 3 Inspection of high consequence components subject to multiple degradation mechanisms	No action required. This item does not apply to components in Westinghouse designed reactors.
SER Section 4.1.4, Topical Report Condition 4 Minimum examination coverage criteria for "expansion" inspection category components	In accordance with SER Section 4.1.4, IPEC will meet the minimum inspection coverage specified in the SER. The appropriate wording has been added to Table 5-3 examination coverage.
SER Section 4.1.5, Topical Report Condition 5 Examination frequencies for baffle-former bolts	In accordance with SER Section 4.1.5, the examination frequency for baffle-former bolts specifies a 10-year inspection frequency following the baseline inspection in Table 5-2.
SER Section 4.1.6, Topical Report Condition 6 Periodicity of the re-examination of "expansion" inspection category components	In accordance with SER Section 4.1.6, Table 5-3 requires a 10-year re-examination interval for all Expansion inspection category components once degradation is identified in the associated Primary inspection category component and examination of the expansion category component commences.
SER Section 4.1.7, Topical Report Condition 7 Updating of industry guideline	No plant-specific action required.
SER Section 4.2.1, Applicant/Licensee Action Item 1	The evaluation of design and operating history demonstrating that MRP-227-A is applicable to IPEC is contained in Section 3.6.
SER Section 4.2.2, Applicant/Licensee Action Item 2	The IPEC review of components within the scope of license renewal against the information contained in MRP-191 Table 4-4 is discussed in Section 3.6.
SER Section 4.2.3, Applicant/Licensee Action Item 3	The IPEC discussion regarding guide tube support pins (split pins) is contained in Section 3.6.
SER Section 4.2.4, Applicant/Licensee Action Item 4	No action required. This item does not apply to Westinghouse designed units.
SER Section 4.2.5, Applicant/Licensee Action Item 5	The IPEC discussion regarding hold down springs is contained in Section 3.6.
SER Section 4.2.6, Applicant/Licensee Action Item 6	No action required. This item does not apply to Westinghouse designed units.
SER Section 4.2.7, Applicant/Licensee Action Item 7	The IPEC discussion regarding lower support column bodies is contained in Section 3.6.
SER Section 4.2.8, Applicant/Licensee Action Item 8	The submittal of information for staff review and approval is discussed in Section 3.6.

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## **6.0 OPERATING EXPERIENCE AND ADDITIONAL CONSIDERATIONS**

### **6.1 Internal and External Operating Experience**

Operating experience related to degradation of reactor internal components covered in this program will be reviewed on a periodic basis. This review will include both domestic and international experience and will be documented in accordance with the Entergy operating experience process. Results of reactor internal components inspected in accordance with MRP-227-A will be collected and summarized in accordance with NEI 03-08 guidelines.

### **6.2 Spring 2016 Operating Experience**

In the spring of 2016, during IP2 outage 2R22, ultrasonic (UT) and/or visual inspections of all 832 baffle former bolts (bolts) were performed in accordance with the NRC approved guidelines in MRP-227-A. Visual inspection of the baffle plates and bolts identified 31 degraded bolts. The UT inspections identified indications on 182 bolts and also determined that 14 bolt locations were not testable. The locations that were not testable were conservatively assumed to possess bolts that failed to meet the acceptance criteria. As a result of the inspection findings, all 227 bolts (31+182+14) with actual and assumed indications were replaced. An additional 51 bolts were replaced to reduce the probability of future failures as well as minimize the probability of clusters of failed bolts. Therefore, during 2R22, a total of 278 bolts (227+51) were replaced.

As a result of the IP2 inspection findings and other industry Operating Experience (OE) indicating a significant number of failed bolts at other similarly-designed PWR plants, the IPEC PWR Vessel Internals Program was revised. In view of the 2R22 inspection findings, Entergy arranged for the fractographic examination of eight baffle former bolts removed from the IP2 baffle structure during the Spring 2016 outage at Westinghouse Electric Company's hot cell laboratory in Churchill, PA. The results of those fractographic examinations are documented in Westinghouse Report MCOE-TR-16-18, Revision 0, "Fractography of Indian Point Unit 2 Baffle Former Bolts" (Nov. 30, 2016). Industry-sponsored metallurgical analysis and materials property testing of additional baffle former bolt specimens from IP2 and other PWRs are still in progress.

Based on as-found conditions and current industry knowledge, including the results of the fractographic examinations of the eight IP2 baffle former bolts discussed in Westinghouse Report MCOE-TR-16-18, IPEC concludes that performing a volumetric examination (i.e., UT) of the required original bolts during each refueling outage, and replacing those bolts found to be degraded until none of the remaining original bolts are required to be credited for the baffle structure to be capable of performing its intended safety function, is a reasonable and acceptable approach. Accordingly, IPEC plans to take the actions specified in paragraphs 1-5 below. These actions are subject to possible revision per the OE program based on the results of ongoing and

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planned future inspection and testing of baffle former bolts from IP2 and other PWR plants. Any findings that result from the following actions will be input to the Corrective Action Program.

1. The IP3 baffle bolt inspections that were previously scheduled to be performed in 3R20 (Spring 2019) will be performed in 3R19 (Spring 2017). Visual and UT inspections on 100% of all accessible baffle former bolts, and a visual inspection of the baffle-edge bolts and baffle former assembly, will be performed in 3R19.
2. Entergy will perform a UT inspection of 100% of the original bolts at IP2 and IP3 during each of the subsequent refueling outages if any of the original bolts are required to remain structurally capable of carrying their design load to ensure structural integrity of the baffle structure during all design conditions.
3. Entergy will also perform a general visual inspection to identify anomalies in the baffle structure at IP2 and IP3 during each subsequent refueling outage.
4. Entergy will perform a UT inspection of inservice replaced (new) bolts if the general visual inspections performed in accordance with paragraph 3. above identify degraded new bolts.
5. Entergy will replace all bolts with indications that are needed to remain structurally capable of carrying their design load to ensure structural integrity of the baffle structure during all design conditions. Additional "good" or anti-cluster bolts will also be replaced to ensure that sufficient margin is maintained to accommodate the same failure rate until the next inspection as the failure rate identified during the current refueling outage. This margin will ensure compliance with the intent of the guidelines provided in WCAP-17096, Revision 2, "Reactor Internals Acceptance Criteria Methodology and Data Requirements."