PSEG Nuclear LLC
P.O. Box 236, Hancocks Bridge, New Jersey 08038-0236

PSEG
Nuclear LLC

AUG 1 0 2018

NEI 03-08

LR-N18-0080

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, DC 20555-0001

Salem Generating Station Unit 1
Renewed Facility Operating License No. DPR-70
NRC Docket No. 50-272

Subject: Notification of Deviation from Electric Power Research Institute (EPRI), "Materials Reliability Program: Pressurized Water Reactor Internals Inspection and Evaluation Guidelines (MRP-227-A)"

PSEG Nuclear LLC (PSEG) is providing notification that Salem Unit 1 has processed a deviation from a Nuclear Energy Institute (NEI) 03-08, "Guideline for the Management of Material Issues", Revision 3, "Needed" work product element in EPRI MRP 227-A, "Pressurized Water Reactor Internals Inspection and Evaluation Guidelines," with appropriate justification and documentation.

NEI 03-08 allows deviation from "Needed" elements with the appropriate justification and documentation. The deviation was approved with the appropriate levels of PSEG management. Enclosure 1 provides the PSEG Materials Degradation Management Process (MDMP) Deviation Form. Enclosure 2 provides the technical evaluation of the Salem Unit 1 Deviation of MRP-227-A Needed Examinations. This notification is provided for information only and no approval or action is expected.

There are no regulatory commitments contained in this letter.

Should you have any questions concerning this matter, please contact Mr. Brian Thomas at 856-339-2022.

Sincerely,

Paul R Duke

Paul R. Duke, Jr
Manager – Licensing
PSEG Nuclear LLC

Enclosure 1: PSEG Materials Degradation Management Process (MDMP) Deviation Form
Enclosure 2: Salem Unit 1 Deviation of MRP-227-A Needed Examinations

cc: Administrator, Region I, NRC
NRC Senior Resident Inspector, Salem
NRC Project Manager, Salem
NRC Project Manager, MRP-227
P. Mulligan, Chief, NJBNE
L. Marabella, Corporate Commitment Tracking Coordinator
T. Cachaza, Salem Commitment Tracking Coordinator

Enclosure 1

PSEG Materials Degradation Management Process (MDMP) Deviation Form

ER-AA-4003
Attachment 2
MDMP Deviation Form

Utility: PSEG Nuclear LLC

Applicable Station(s) and Unit No.: Salem Unit 1

Utility Contact(s): Patrick Fabian, Corporate Engineering Programs
Tim Giles, Salem Engineering Programs

Issue Program (IP) activity or document: Materials Reliability Program: Pressurized Water Reactor Internals Inspection and Evaluation Guidelines (MRP-227-A)

Scope / Description of Deviation:

PSEG Nuclear LLC (PSEG) is deferring three normally inaccessible Pressurized Water Reactor (PWR) Vessel Internals Program primary component inspections for one refueling outage from Salem Unit 1 refueling outage 1R26 in 2019 to refueling outage 1R27 in 2020. The deferral aligns inspections for components only accessible with the core barrel (CB) removed for the PWR Vessel Internals Program and American Society of Mechanical Engineers (ASME) Section XI. In January 2018, baffle-former bolt expansion inspection interim guidance was issued by Electric Power Research Institute (EPRI) Materials Reliability Program (MRP) interim guidance letter MRP 2018-002. This interim guidance supplements the current expansion criteria for baffle-former bolts as documented in MRP-227-A, requiring inspection of barrel-former bolts within three fuel cycles after initial discovery of significant baffle-former bolt clustering. PSEG identified significant baffle-former bolt clustering in Salem Unit 1 refueling outage 1R24, and therefore PSEG is required to complete barrel-former bolt inspections by refueling outage 1R27. Barrel-former bolt inspections require the CB to be removed from the reactor vessel for access to the bolts. PSEG originally planned PWR Vessel Internals Program and ASME Section XI inspections for refueling outage 1R26 with the CB removed. Deferring the removal of the CB to refueling outage 1R27 provides PSEG time to prepare for the required barrel-former bolt expansion inspections, aligns inspections of components requiring the CB removed, eliminates risks of removing the CB more than once, and reduces radiological dose to plant workers.

Inspections of components accessible with the CB installed will be completed during refueling outage 1R26; however, some components cannot be accessed during refueling outage 1R26 while the CB is installed. Per the requirements of MRP-227-A the "needed" examinations for these components shall be completed no later than two refueling outages from the beginning of the license renewal period. Refueling outage 1R26 is the second refueling outage after entry into the period of extended operation in August 2016 (Cycle 25 started July 30, 2016). Based on the industry operating experience gained while performing MRP-227-A examinations, and the evaluation provided in SAP Order 70201108, PSEG has determined the acceptability of deferring MRP-227-A "needed" examinations for the lower core barrel cylinder girth weld, lower core barrel flange weld, and thermal shield flexures by one refueling cycle (from outage 1R26 in 2019 to outage 1R27 in 2020).

ER-AA-4003
Attachment 2
MDMP Deviation Form

Reason for Deviation:

Deferring the removal of the core barrel to refueling outage 1R27 provides PSEG time to prepare for the required MRP-227 barrel-former bolt expansion examinations, aligns examinations of components requiring the core barrel removed, eliminates risks of removing the core barrel more than once, and reduces radiological dose to plant workers.

Technical Justification for Deviation: SAP Notification 20800427, SAP Order 70201108

Time Frame the Deviation will be in Effect: This deviation will be in effect until Salem Unit 1 refueling outage 1R27 (Fall 2020).

Deviation to this IP document is classified as: NEEDED

The MRP-227-A section 7.3 "needed" implementation requirement states: *"Each commercial U.S. PWR unit shall implement Tables 4-1 through 4-9 and Tables 5-1 through 5-3 for the applicable design within twenty-four months following issuance of MRP-227-A".*

Prepared by:	Patrick Fabian	[signature]	Date: 8-1-2018
Salem Programs Manager:	Mike Ambrosino	[signature]	Date: 8-1-18
Corporate Programs Manager:	Ali Fakhar	[signature]	Date: 8/1/2018
Salem Engineering Director:	Rob DeNight	[signature]	Date: 8/5/18
Engineering Services Director:	Len Rajkowski	[signature]	Date: 8/6/18
Salem Station VP:	Chaz McFeaters	[signature]	Date: 8/6/18
Nuclear Engineering VP:	Paul Davison	[signature]	Date: 8/6/18

Enclosure 2

Salem Unit 1

Deviation of MRP-227-A Needed Examinations

1. Reason for Evaluation/Scope

PSEG Nuclear LLC (PSEG) is deferring three normally inaccessible Pressurized Water Reactor (PWR) Vessel Internals Program primary component inspections for one refueling outage from Salem Unit 1 refueling outage 1R26 in 2019 to refueling outage 1R27 in 2020. The deferral aligns inspections for components only accessible with the core barrel (CB) removed for the PWR Vessel Internals Program and American Society of Mechanical Engineers (ASME) Section XI. In January 2018, baffle-former bolt expansion inspection interim guidance was issued by Electric Power Research Institute (EPRI) Materials Reliability Program (MRP) interim guidance letter MRP 2018-002 [4.3]. This interim guidance supplements the current expansion criteria for baffle-former bolts as documented in MRP-227-A [4.2], requiring inspection of barrel-former bolts within three fuel cycles after initial discovery of significant baffle-former bolt clustering. PSEG identified significant baffle-former bolt clustering in Salem Unit 1 refueling outage 1R24, and therefore PSEG is required to complete barrel-former bolt inspections by refueling outage 1R27. Barrel-former bolt inspections require the CB to be removed from the reactor vessel for access to the bolts. PSEG originally planned PWR Vessel Internals Program and ASME Section XI inspections for refueling outage 1R26 with the CB removed. Deferring the removal of the CB to refueling outage 1R27 provides PSEG time to prepare for the required barrel-former bolt expansion inspections, aligns inspections of components requiring the CB removed, eliminates risks of removing the CB more than once, and reduces radiological dose to plant workers.

Inspections of components accessible with the CB installed will be completed during refueling outage 1R26; however, some components cannot be accessed during refueling outage 1R26 while the CB is installed. Per the requirements of MRP-227-A the "needed" examinations for these components shall be completed no later than two refueling outages from the beginning of the license renewal period. Refueling outage 1R26 is the second refueling outage after entry into the period of extended operation in August 2016 (Cycle 25 started July 30, 2016 [4.15]). Based on the industry operating experience gained while performing MRP-227-A examinations, and the evaluation provided below, PSEG has determined the acceptability of deferring MRP-227-A "needed" examinations for the lower core barrel cylinder girth weld, lower core barrel flange weld, and thermal shield flexures by one refueling cycle (from outage 1R26 in 2019 to outage 1R27 in 2020). See Attachment 1 and 2 for illustration of these components.

The purpose of this evaluation is to document technical justification for deviation from MRP-227-A needed requirements, consistent with PSEG procedures ER-AA-4003 [4.8] and CC-AA-309-101 [4.9], and NEI 03-08 [4.1]. The detailed evaluation provided below is applicable to Salem Unit 1.

2. Detailed Evaluation

The PWR Vessel Internals Program manages the effects of age-related degradation mechanisms of reactor vessel internals. The program is applicable to Salem Unit 1 and Salem Unit 2, administratively managed by PSEG procedure [4.7] and required to comply with EPRI MRP-227-A, "Materials Reliability Program: PWR Internals Inspection and Evaluation Guidelines". The Materials Reliability Program (MRP) developed the inspection and evaluation (I&E) guidelines for managing long-term aging of reactor vessel internal components of pressurized water reactors. Specifically, the guidelines are applicable to reactor vessel internal structural components; they do not address fuel assemblies, reactivity control assemblies, or welded attachments to the reactor vessel.

The I&E guidelines were created with the premise that component aging degradation can be managed by selection and monitoring of lead components and locations for the applicable aging degradation mechanisms. These lead components are designated as "Primary" in the I&E guidelines. The primary components for Westinghouse plants listed in Table 4-3 of MRP-227-A that are applicable to Salem Unit 1 are as follows:

- Control rod guide tube (CRGT) assembly: guide cards
- CRGT assembly: lower flange welds
- Upper core barrel flange weld
- Upper core barrel cylinder girth weld
- Lower core barrel cylinder girth weld
- Lower core barrel flange weld
- Baffle-edge bolts
- Baffle-former bolts
- Baffle-former assembly
- Internals hold down spring
- Thermal shield flexures

Depending on the inspection results of the primary components, expansion components may need to also be inspected, and these expansion components are prescribed in MRP-227-A, Table 4-6. The PSEG PWR Vessel Internals program represents a Salem Station License Renewal Commitment (CM-SC-2009-626) included in Salem UFSAR, Appendix B, A.2.1.7, PWR Vessel Internals [4.10]. The commitment includes the following:

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.
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 inspection plans, also referred to as Reactor Internals Aging Management Program Plan (AMP), are documented in WCAP-17397 [4.13] and WCAP-17438 [4.14]. These plans were submitted to the NRC [4.11] consistent with the commitment, and approved by NRC Staff Assessment [4.12]. In summary, the PSEG AMPs submitted to the NRC affirmed PSEG would follow the Inspection and Evaluation (I&E) Guidelines of MRP-227-A.

MRP-227-A is a 'work product' of the EPRI MRP, an 'Issue Program (IP)' as defined in NEI 03-08. Appendix B to NEI 03-08, Implementation Protocol, defines the processes and expectations for implementing industry guidance issued under the Materials Initiative, and requires that IPs identify the specific implementation category for 'requirements' identified by guideline-type work products.

The three implementation categories described in NEI 03-08 are as follows:

- Mandatory – to be implemented at all plants where applicable;
- Needed – to be implemented wherever possible, but alternative approaches are acceptable; and
- Good Practice – implementation is expected to provide significant operational and reliability benefits, but the extent of use is at the discretion of the individual utility.

MRP-227-A lists one mandatory and several needed implementation requirements. The mandatory requirement is to develop and document a program for management of aging of reactor internal components. The needed requirements include implementation of Tables 4-1 through 4-9 and Tables 5-1 through 5-3 for the applicable plant design (section 7.3 of MRP-227-A).

Table 4-3 provides the primary inspection components for Westinghouse designed plants, applicable to Salem Unit 1 (and Salem Unit 2). Table 4-3 includes the examination requirements for the lower core barrel cylinder girth weld, lower core barrel flange weld, and thermal shield flexures. The inspection requirements for these components prescribed in Table 4-3 are enhanced visual (EVT-1) examination for the CB welds and visual (VT-3) for the thermal shield flexures, no later than 2 refueling outages from the beginning of the license renewal period. 1R26 (2019) is the second refueling outage after entry into the period of extended operation in August 2016. Inspection of the core barrel cylinder girth weld, lower core barrel flange weld, and thermal shield flexures was originally planned for outage 1R26, and is being deferred one refuel cycle (outage 1R27).

PSEG has already completed some of the MRP-227-A primary component examination requirements, in outage 1R24 (2016) and 1R25 (2017), summarized below.

Completed MRP-227-A Inspections in 1R24 (Spring 2016)	
Component	**Exam Type**
Baffle-Former Assembly: Baffle-edge bolts	VT-3
Baffle-Former Assembly: Baffle-former bolts	UT
Baffle-Former Assembly: Assembly (Includes: Baffle plates, baffle edge bolts, and indirect effects of void swelling in former plates)	VT-3

Completed MRP-227-A Inspections in 1R25 (Fall 2017)	
Component	**Exam Type**
Control Rod Guide Tube Assembly: Guide plates (cards)	VT-3

PSEG is planning to complete further MRP-227-A primary component examination requirements in 1R26 (2019).

Planned MRP-227-A Inspections in 1R26 (Spring 2019)	
Component	**Exam Type**
Control Rod Guide Tube Assembly: Lower flange welds	EVT-1
Core Barrel Assembly: Upper core barrel flange weld	EVT-1
Core Barrel Assembly: Upper core barrel cylinder girth weld	EVT-1

The remaining Salem Unit 1 MRP-227-A primary component examinations originally scheduled for 1R26 (2019), which are being deferred until 1R27 (2020), are listed below.

Deferred MRP-227-A Inspections to 1R27 (Fall 2020)	
Component	**Exam Type**
Core Barrel Assembly: Lower core barrel cylinder girth weld	EVT-1
Core Barrel Assembly: Lower core barrel flange weld	EVT-1
Thermal Shield Assembly: Thermal shield flexures	VT-3

> NOTE: The Primary component *Alignment and Interfacing Components:* Internals hold down spring does not require examination (measurement of spring height) until 1R27 (2020); and is therefore not addressed in this evaluation.

The subsections below evaluate the primary inspection components being deferred which require evaluation for the deviation to MRP-227-A.

2.1 Lower Core Barrel Cylinder Girth Weld and Lower Core Barrel Flange Weld

The technical basis for the planned deviation from the MRP-227-A "Needed" requirements for the lower CB cylinder girth weld and the lower CB flange weld is provided from Westinghouse LTR-AMLR-18-47 [4.16] and summarized in this section. The technical basis presented here is based on a combination of three factors: industry operating experience (OE), acceptable crack length, and CB functionality.

Core Barrel Weld Operating Experience

Per MRP-227-A, both the lower CB cylinder girth weld and lower CB flange weld have a "Needed" requirement to perform an enhanced visual EVT-1 examination of 100% of one side of the accessible surfaces of the weld and adjacent base metal. Inspections of these two welds and other similar CB welds under MRP-227-A have been completed at multiple Westinghouse-designed plants. These inspections are intended to identify crack-like surface flaws in the welds or adjacent base metal. The inspection results are reported in Materials Reliability Program biennial reports [4.4] [4.5] [4.6].

Core Barrel Assembly: Lower Core Barrel Cylinder Girth Weld

EVT-1 inspections under MRP-227-A have been conducted on the lower CB cylinder girth weld for the following Westinghouse-designed plants: D.C. Cook Unit 1, Farley Unit 1, Indian Point Unit 2, Point Beach Unit 1, Point Beach Unit 2, Prairie Island Unit 1, Prairie Island Unit 2, H.B. Robinson, Surry Unit 1, Turkey Point Unit 3, and Turkey Point Unit 4. These inspections were conducted after these plants had operated between a minimum of 29.1 effective full-power years (EFPY) and a maximum of 35.5 EFPY with an average of 32.2 EFPY. No service-related degradation or recordable indications were detected at any of these units. Salem Unit 1 will accumulate approximately 29.5 EFPY at the time of refueling outage 1R26, and 30.9 EFPY at refueling outage 1R27 [4.15] both of which are within the range of EFPY for the inspections in these other plants.

Core Barrel Assembly: Lower Core Barrel Flange Weld

EVT-1 inspections under MRP-227-A have been conducted on the lower CB flange weld for the following Westinghouse-designed plants: D.C. Cook Unit 1, Farley Unit 1, Indian Point Unit 2, Point Beach Unit 1, Point Beach Unit 2, Prairie Island Unit 1, Prairie Island Unit 2, H.B. Robinson, Surry Unit 1, Turkey Point Unit 3, and Turkey Point Unit 4. These inspections were conducted after these plants had operated between a minimum of 29.1 EFPY and a maximum of 35.5 EFPY with an average of 32.2 EFPY. No service-related degradation or recordable indications were detected at any of these units. Salem Unit 1 will accumulate approximately 29.5 EFPY at the time of refueling outage 1R26, and 30.9 EFPY at refueling outage 1R27 [4.15] both of which are within the range of EFPY for the inspections in these other plants.

Other Relevant Westinghouse-Designed Core Barrel Weld Inspections

Inspections of the other Primary components from MRP-227-A also provide valuable information for understanding the likely behavior of the lower CB flange weld. Both the upper CB flange weld and the upper CB cylinder girth weld are subject to cracking degradation effects relevant to the lower CB flange weld—stress corrosion cracking (SCC) for both welds and fatigue for the upper CB cylinder girth weld. The welds at these different locations in the CB are similar in material and geometry, so this comparison can be made. Note that the lower CB cylinder girth weld is subject to irradiation-assisted SCC (IASCC), which differentiates it from these other welds and makes such a comparison less valid.

EVT-1 inspections under MRP-227-A have been conducted on the upper CB flange weld for the following Westinghouse-designed plants: D.C. Cook Unit 1, Farley Unit 1, R.E. Ginna, Indian Point Unit 2, North Anna Unit 1, Point Beach Unit 1, Point Beach Unit 2, Prairie Island Unit 1, Prairie Island Unit 2, H.B. Robinson, Surry Unit 1, Surry Unit 2, Turkey Point Unit 3, and Turkey Point Unit 4. No service-related degradation or recordable indications were detected at any of these units.

EVT-1 inspections under MRP-227-A have been conducted on the upper CB cylinder girth weld for the following Westinghouse-designed plants: D.C. Cook Unit 1, Farley Unit 1, Indian Point Unit 2, North Anna Unit 1, Point Beach Unit 1, Point Beach Unit 2, Prairie Island Unit 1, Prairie Island Unit 2, H.B. Robinson, Surry Unit 1, Turkey Point Unit 3, and Turkey Point Unit 4. No service-related degradation or recordable indications were detected at any of these units.

Other Relevant Operating Experience

The core support barrel in a CE-designed PWR is similar in design to the CB in Westinghouse-designed plants. MRP-227-A contains "Needed" requirements for the core support barrel girth welds with the same inspection type (EVT-1), coverage, and timing as the Westinghouse CB welds. Core support barrel welds have been inspected at several CE-designed plants. No service-related degradation or recordable indications were detected in all but one of those units.

In spring 2018, one CE-designed unit performed inspections of the core support barrel lower cylinder girth welds, specifically the middle girth weld [4.6]. The reported operating time for that plant at the time of inspection was 33.65 EFPY. The middle girth weld in a CE core support barrel is located in the core beltline region near the fuel. One small, crack-like indication approximately 1.3 inches in length was detected adjacent to the middle girth weld. During the same outage, supplemental visual and UT inspections were also conducted on the core support barrel middle axial weld, which is included in MRP-227-A as an Expansion component. The middle axial weld is located above the middle girth weld in a CE-designed core support barrel, intersects the middle girth weld, and runs perpendicular to the girth weld. These supplemental inspections detected 45 additional crack-like indications located adjacent to the middle axial weld. Most of the indications along the middle axial weld were oriented perpendicular to the weld (i.e., circumferential to the core support barrel cylinder). The largest of these indications was 1.88 inches in length, which is less than 2% of the circumference of the core support barrel, even after applying flaw proximity rules. The plant has returned to service for one additional cycle of operation based on application of flaw acceptance criteria.

The root cause of these crack-like indications by the middle axial and middle girth welds is not currently known. This CE-designed plant did experience significant cracking degradation of the core support barrel early in life due to a failure of the plant's thermal shield [4.2]. The degradation was mitigated by removing the thermal shield and blunting the existing cracks. It is possible that the degradation observed in 2018 was a result of this early cracking that was not detected until the application of the enhanced visual inspection techniques required by MRP-227-A. It is also possible that the degradation is a result of the IASCC degradation mechanism that was screened in for both of these welds. The OE for the middle girth weld and middle axial weld is most applicable to the lower CB cylinder girth weld at Salem Unit 1 because of the similarities in design and the weld's location near the core. The operating life of the CE plant is slightly higher than the projected operating time of Salem Unit 1 for both 1R26 and 1R27 (33.65 EFPY as compared to 29.5 or 30.9 EFPY [4.15]), so Salem Unit 1 should have less potential for experiencing cracking of the core barrel welds by the completion of those two outages.

Conclusion:

The OE for inspections relevant to the lower CB cylinder girth weld and lower CB flange weld has generally been favorable. Inspections of these two welds in Westinghouse plants have produced positive results—no service-related degradation or relevant indications have been detected in multiple MRP-227-A EVT-1 inspections. Inspections of similar welds in Westinghouse plants and several CE-designed plants have also all been favorable. Inspection experience from one CE-designed plant during spring 2018 resulted in the observation of multiple crack-like indications at the middle girth and middle axial welds. This experience is most applicable to the lower CB cylinder girth weld at Salem Unit 1 because it is also located near the core. Deferring the MRP-227-A Primary inspections for the lower CB cylinder girth weld and lower CB flange weld is generally supported by the favorable inspection experience from multiple plants. However, the recent CE plant OE requires some additional technical basis for deferral of the inspection by one cycle at Salem Unit 1. This technical discussion is contained in the sections below.

Core Barrel Crack Length Acceptance Criteria

Based on the OE presented above, most plants have not observed crack-like indications during the detailed MRP-227-A inspections, and in the one case where indications were observed, they were limited in size. The plant that observed crack-like indications during the Spring 2018 outage season was able to disposition all of the observed flaws and continue operation for one additional cycle. This limited time of continued operation was governed by the requirement within WCAP-17096-NP-A [4.17] to verify the crack growth rate used in the acceptance criteria evaluation. Per the acceptance criteria methodology, "The flaw evaluation will address the verification process of the predicted crack growth rate. Depending on the flaw size and knowledge of the plant conditions, a re-inspection at the next refueling outage may be required to provide data needed to operate beyond one cycle." With verification of the crack growth rate, this period of continued operation could have been longer.

Per the functionality evaluations described in PWROG-17084-P [4.18], circumferential cracking is bounding for impact on the functionality of the CB. A circumferential crack can potentially result in a fully severed and dropped section of the CB, while an axial crack has to become large and wide open before it can impact safety through bypass flow.

Flaw tolerance acceptance criteria have been calculated for the lower CB cylinder girth weld and lower CB flange weld at Salem Unit 1 according to the NRC-approved methodologies of WCAP-17096-NP-A [4.17]. The allowable crack length for an 18-month return to service for the lower CB cylinder girth weld is greater than 40 inches and the lower CB flange weld is greater than 50 inches [4.19]. These allowable crack lengths are quite generous, permitting a crack that is approximately 9-12% of the CB circumference (circumference of the Salem Unit 1 CB is approximately 470 inches [4.20]). By comparison, the largest of the crack-like indications observed at the CE-designed plant in Spring 2018 was 1.88 inches long, which is less than 2% of the core support barrel circumference in length. Even if it is assumed that in 2019 Salem Unit 1 has cracking with a similar number and size of flaws, at least one more cycle of operation could be achieved. This conclusion is based on the facts that the CE-designed plant was able to continue operation for one cycle with that level of cracking and that Salem Unit 1 has significant margin beyond this assumed flaw size. Using the 10-year and 18-month allowable crack lengths provided for Salem Unit 1 in WCAP-18070-P [4.19], the amount of crack growth expected over that 8.5 year period can be calculated, which is conservative relative to the 18-month delay planned for Salem Unit 1. The largest crack growth is in the lower core barrel cylinder girth weld and is only predicted at 0.1 inches over 8.5 years. Over an 18-month period, this small amount of crack growth would not cause flaws like those seen at the CE-designed plant to exceed the allowable circumferential flaw sizes for an 18-month return to service. Even if one conservatively assumes that in 2019 Salem Unit 1 will have flaws three or four times larger than those observed at the CE-designed plant in spring 2018, the plant could still continue operation for another 18 months.

Based on these observations and the generous allowable crack lengths calculated for the two welds, it is concluded that operation for one additional cycle at Salem Unit 1 before performing the MRP-227-A inspections of the welds is acceptable. There is adequate margin in the allowable crack length and crack growth to ensure continued functionality during the one cycle delay in the initial inspection of these welds.

Functionality Considerations for the Safety Impacts of CB Degradation

The functional impact of CB degradation due to cracking effects has been evaluated in responses to U.S. Nuclear Regulatory Commission (NRC) requests for additional information (RAIs) on MRP-227, Revision 1 [4.21] [4.22] [4.23] and a recent Pressurized Water Reactor Owners Group (PWROG) project [4.18]. According to WCAP-17096-NP-A [4.17], the function of the core barrel girth welds is to act as a primary core support structure. The pressurized water reactor issue management table in MRP-156 [4.24] provides additional detail on the core barrel functions:

- Provides core support through other internals structures attached to the barrel, such as the lower core plate and baffle-former assembly
- Directs coolant flow to the core and out of the vessel
- Maintains the capability to insert the controls for safe shutdown through the lower core plate and providing alignment for the upper core plate

These reference documents [4.18] [4.22] divided the evaluation of CB function into effects during a **faulted event** and effects during **normal operation**. The same division will be used here.

Faulted Event

Faulted events include occurrences such as large seismic events or loss of coolant accidents and will trigger an automatic insertion of the core control elements (control rods) [4.18]. The CB is designed to withstand such events. However, if a circumferentially-oriented flaw longer than the critical crack length is present in the barrel, then it is possible that the flaw could propagate and result in either a partially or fully separated portion of the barrel. If such a separation occurs, the safe shutdown functionality of the CB is maintained by the secondary core support (SCS) system.

For Salem Unit 1, the SCS system relies on the SCS structure, the radial keys and clevises, the fuel alignment pins on the upper and lower core plates, the CB outlet nozzles, and the reactor vessel [4.18] [4.22] [4.23]. The SCS structure is an assembly of columns and plates at the bottom of the core barrel that only allows a severed barrel to drop a short distance before it catches the barrel on the bottom of the reactor vessel. The SCS assembly includes energy absorption columns to reduce the impact of the hypothetical drop. If the barrel drops, the radial keys and clevises and the fuel alignment pins would still be engaged, due to the limited drop distance permitted by the SCS structure. The radial keys and clevises and CB outlet nozzles prevent unacceptable tilting or rotation of the barrel section relative to the upper internals assembly and the control rods. The fuel alignment pins maintain the alignment of the control rod assemblies with the individual fuel assemblies, which allows control rod insertion for safe shutdown.

In the case of a partial circumferential failure of the CB, the SCS system will still effectively meet these functions to ensure safe shutdown of the reactor [4.18] [4.22] [4.23]. The SCS structure limits the potential drop on the fractured side of the barrel, and the radial keys, clevises, fuel alignment pins, and CB outlet nozzles will maintain the alignment of the core to allow insertion of the control rods.

Full or partial circumferential separation of the barrel at different locations along its axis will result in some differences in how the barrel is aligned and supported, but in all cases, portions of the SCS system will remain engaged to ensure functionality [4.18]. For example, the CB outlet nozzles will contribute less to the alignment of a failed barrel if the failure occurs below the nozzles [4.23]. Both the lower CB cylinder girth weld and lower CB flange weld are located below the nozzles. Alignment of the core relative to the control rods would still be maintained by the clevis inserts, radial keys, and fuel alignment pins. The CB outlet nozzles will continue to support the upper portion of the barrel that remained in place.

Safe shutdown relies on the ability to insert the control rods when needed. For Westinghouse-designed plants, like Salem Unit 1, testing was conducted to determine the impact of a core drop on control rod insertion times [4.18] [4.23]. Two relevant tests were conducted—the first simulated the lateral offset that would occur between the fuel assembly top nozzle and the control rod insertion path and the second simulated a lateral offset of the fuel assembly at mid span with the top and bottom of the fuel assembly pinned. The first case resulted in less than an 8% increase in rod insertion times and the second in less than a 2% impact.

Once safe shutdown has been achieved, the long-term ability to remove the decay heat from the core should also be considered. Degradation of the core barrel causes increased bypass flow at a location dependent on the location of the failure [4.18], but the limited distance that the core barrel can drop, due to the secondary core support structure, keeps the fuel alignment pins, clevis inserts and radial keys, and outlet nozzles engaged. This short drop will not impede the flow of coolant into or out of the vessel through the inlet and outlet nozzles or isolate the core from continued coolant flow. Therefore, a failure of the core barrel will not impede the ability to keep the core fully covered with coolant after a faulted event, ensuring long-term heat removal.

If a faulted event occurs at Salem Unit 1 between the 1R26 and 1R27 outages, the event would result in a plant shutdown and corrective actions to address potential effects. Failure of the core barrel is only possible if a flaw larger than the allowable crack lengths, presented in prior discussions in section "Core Barrel Crack Length Acceptance Criteria", is present in the CB. Faulted events have a low probability of occurrence due to design and, based on the preceding discussions, a flaw larger than the acceptable crack size is also a low probability event. Such a coincidence of events is unlikely, and even if it does occur, the plant will be able to achieve safe shutdown.

Normal Operation

If the core barrel failed during normal or upset operating conditions and was not immediately shut down, the resultant gap could cause several effects. The nature of the effects would depend on the failure location [4.18]. Only the lower CB cylinder girth weld and the lower CB flange weld are being deferred for one cycle at Salem Unit 1. As noted above for the faulted condition, circumferential failure at these locations will be bounding. For safe shutdown considerations, the details provided on the support and alignment provided by the SCS system are also applicable for normal operation. If a complete or partial separation of the core barrel occurs, the interface between the fuel and the control rod assemblies will be maintained, permitting safe shutdown.

The difference from the faulted case for a failure that occurs during normal or upset operating conditions is that it may not be automatically detected and lead to a plant shutdown immediately [4.18] [4.22]. The impact of a drop is expected to register on seismic detection systems and loose parts monitoring systems, but based on current knowledge, may not lead to an automatic shutdown.

A failure in the lower CB cylinder girth weld would occur at the same elevation as the baffle-former assembly and the thermal shield [4.18]. The design of the former plates attached to the inner diameter of the CB and the baffle plates is expected to allow the assembly to accommodate the separation associated with a failure and CB drop event. Some distortion of the assembly will occur and can impact the fuel, but lateral support of the fuel will be maintained. The thermal shield lower flexures would likely yield significantly or fail completely, but the upper supports would continue to hold the thermal shield.

The gap due to this hypothetical failure of the lower CB cylinder girth weld could also lead to increased bypass flow around the core [4.18]. Inside of the barrel, the baffle-former assembly may be flexible enough to accommodate the axial drop of the CB while still limiting the amount of bypass flow. If the damage to the baffle-former assembly is more severe, additional bypass flow and flow impingement on the core could occur. Outside of the barrel, the presence of the thermal shield is expected to reduce the amount of bypass flow that could occur through the gap in the failed CB.

The effects of a failure in the lower CB flange weld would be less severe than those for the lower CB cylinder girth weld [4.18]. A hypothetical failure at the flange weld would leave most of the lower internals intact. The SCS system could be compromised and the lower support columns would be subjected to additional deadweight loads and horizontal flow loads from the resulting gap. A failure at the lower flange location could result in failure of the lower support column bolts and could impact the lower core plate. Failure at the lower CB flange weld would leave the fuel only supported by the lower core plate, but the potential tilting caused by this condition has already been accounted for in the control rod drop testing discussed in the faulted event section.

There are multiple monitoring systems in place that may detect the effects of a CB failure during normal or upset operating conditions and result in an automatic or operator-enacted plant shutdown [4.18] [4.22]. As noted earlier, the plant seismic and loose parts detection systems will register the failure event but may not actuate an automatic response. For changes in core bypass flow, the temperature and neutronics monitoring (in-core or ex-core) are likely to detect significant changes, but once again may not result in an automatic response. Finally, if the baffle-former assembly has impinged on the fuel assemblies or if the lower core plate has deformed, the effects should be detected either through reactor coolant system radioactive isotope monitoring, fuel loading and unloading operations, or possibly foreign material exclusion inspections.

This uncertainty in the detectability of a full failure event during normal or upset operation requires additional justification to support operation for a cycle beyond the MRP-227-A inspection timing requirement. The OE described previously provides part of this justification.

Barrel degradation has only been observed at one CE-designed plant and the length of the flaws observed was less than 2% of the core support barrel weld circumference. The acceptance criteria provided in section "Core Barrel Crack Length Acceptance Criteria" were developed considering a faulted event combining seismic and loss of coolant accident effects. The stresses and stress intensities developed during a faulted event are significantly higher than those present in either of these CB welds during normal or upset operating conditions. Thus, the critical crack length for normal and upset operating conditions will be higher than the 9-12% of the CB circumference.

Under normal and upset operating conditions, the flaw tolerance of the CB (as evidenced by the large critical crack length) provides assurance that complete failure of the CB at either the lower CB cylinder girth weld or lower CB flange weld is unlikely. The small size of the crack-like indications observed at the single plant that has experienced CB weld OE supports this conclusion, as does the large number of plants that have not observed any degradation of the CB welds during MRP-227-A inspections. There is adequate margin in the allowable crack length and crack growth to ensure continued functionality during the one cycle delay in the initial inspection of these welds. Furthermore, the effects of a hypothetical CB failure during normal or upset conditions will likely have impacts detectable by various monitoring systems, which may also result in plant shutdown to address the problem.

Summary and Conclusions of Section 2.1

Inspections of the lower CB cylinder girth weld and the lower CB flange weld at Salem Unit 1 are required by cycle 1R26 (2019) under the requirements of MRP-227-A. The technical basis for this deviation includes the following areas:

- Review of the relevant OE for MRP-227-A inspections of CB welds showing that only one plant has observed crack-like indications and that these were limited in size

- Consideration of the acceptable crack length calculated for Salem Unit 1 concluding that the generous critical crack size provides substantial margin relative to observed crack sizes for extending operation by one cycle

- Evaluation of the functional effects of a postulated circumferential CB failure showing that safe shutdown is maintained for both faulted and normal or upset conditions and that even though stopping operation may not be automatic during normal or upset conditions, the critical crack size provides substantial margin for extending operation by one cycle

Thus, it is concluded that it is acceptable for Salem Unit 1 to postpone the inspection of the lower CB cylinder girth weld and the lower CB flange weld for one cycle beyond the required MRP-227-A timing.

2.2 Thermal Shield Flexures

The MRP-227-A examination requirement for the thermal shield assembly: thermal shield flexures is a visual (VT-3) examination no later than two refueling outages from the beginning of the license renewal period and subsequent examinations on a ten year interval. This examination is intended to identify cracking (fatigue) or loss of material (wear) that results in excessive wear, fracture, or complete separation of the thermal shield flexures. Operating experience summaries from Westinghouse designed plants that have completed MRP-227-A examinations of the thermal shield flexures are available in Materials Reliability Program biennial reports [4.4] [4.5] [4.6]. In summary, 12 plants similar to Salem Unit 1 have completed thermal shield flexures inspections with no recordable indications identified. These Westinghouse designed plants are Beaver Valley Unit 1, D.C Cook Unit 1, Ginna, H.B. Robinson, Indian Point 2, Point Beach Unit 1, Point Beach Unit 2, Prairie Island Unit 1, Prairie Island Unit 2, Surry Unit 1, Turkey Point Unit 3, Turkey Point Unit 4. All of the inspection findings at these sites allowed for a ten year re-inspection interval. These inspections were conducted after these plants had operated between a minimum of 29.1 effective full-power years (EFPY) and a maximum of 35.5 EFPY with an average of 32.2 EFPY. Salem Unit 1 will accumulate approximately 29.5 EFPY at the time of refueling outage 1R26, and 30.9 EFPY at refueling outage 1R27 [4.15] both of which are within the range of EFPY for the inspections in these other plants. Therefore there is reasonable assurance deferring the thermal shield flexures inspections until 1R27 is acceptable.

3. Conclusions / Findings

Deferring the Salem Unit 1 MRP-227-A "needed" examinations by one refueling outage is acceptable. There have been no significant discoveries from similar Westinghouse plants that have performed the MRP-227-A inspections. Based on industry operating experience, the short duration of the deferral (one cycle), and the evaluations presented above there are no significant concerns with completing the Salem Unit 1 MRP-227-A examinations for the lower core barrel cylinder girth weld, lower core barrel flange weld, and thermal shield flexures in 1R27 (2020).

4. References

4.1 NEI 03-08, Guidelines for the Management of Materials Issues, Revision 3, February, 2017

4.2 MRP-227-A, Materials Reliability Program: Pressurized Water Reactor Internals Inspection and Evaluation Guidelines, 2011

4.3 MRP 2018-002, Transmittal of NEI-03-08 "Needed" Interim Guidance Regarding MRP-227-A and MRP-227, Revision 1 Baffle-Former Bolt Expansion Inspection Requirements for PWR Plants, January 17, 2018

4.4 MRP 2014-009, Biennial Report of Recent MRP-227-A Reactor Internals Inspection Results, May 12, 2014

4.5 MRP 2016-008, Biennial Report of MRP-227-A Reactor Internals Inspection Results, May 18, 2016

4.6 MRP 2018-025, 2018 Biennial Report of Recent MRP-227-A Reactor Internals Inspection Results, July 19, 2018

4.7 ER-AP-333, Pressurized Water Reactor Vessel Internals Aging Management Program, Revision 1

4.8 ER-AA-4003, Materials Degradation Management Program (MDMP) Deviation Guidance, Revision 3

4.9 CC-AA-309-101, Engineering Technical Evaluations, Revision 11

4.10 Salem UFSAR Appendix B, section A.2.1.7, PWR Vessel Internals Program, Salem Station License Renewal Commitment (CM-SC-2009-626)

4.11 LR-N14-0183, Submittal of PWR Vessel Internals Inspection Plans for Aging Management of Reactor Internals at Salem Generating Station, Units 1 and 2, August 11, 2014 (ML14224A667)

4.12 ML17320A859, Salem Nuclear Generating Station, Unit NOS. 1 and 2 – Staff Assessment of the Reactor Vessel Internals Aging Management Program (CAC NOS. MF5149 and MF5150; EPID L-2014-LRL-0001), November 21, 2017

4.13 WCAP-17397, PWR Vessel Internals Program Plan for Aging Management of Reactor Internals at Salem Nuclear Generating Station, Salem Unit 1, July 2014 (PSEG VTD 902381 sheet 1)

4.14 WCAP-17438, PWR Vessel Internals Program Plan for Aging Management of Reactor Internals at Salem Nuclear Generating Station, Salem Unit 2, July 2014 (PSEG VTD 902382 sheet 1)

4.15 NFS18-052, Salem Unit 1 Cycle 27 Reload Design Initialization, July 2018. *Note: Page 85 provides actual and predicted cycle EFPY from cycle 1 to cycle 31.*

4.16 LTR-AMLR-18-47 Revision 0, Technical Basis Supporting Deviation from Two Core Barrel Girth Weld NEI 03-08 "Needed" Requirements of MRP-227-A for Salem Unit 1, July 27, 2018

4.17 Pressurized Water Reactor Owners Group Report, WCAP-17096-NP-A, Rev 2, Reactor Internals Acceptance Criteria Methodology and Data Requirements, August 2016. (ML16279A320)

4.18 Pressurized Water Reactor Owners Group Report, PWROG-17084-P, Revision 0, Core Barrel Girth and Flange Weld Aging Management Strategy (Probability and Consequences of Failure), June 2018.

4.19 Westinghouse Report, WCAP-18070-P, Revision 0, Background and Technical Basis Supporting Engineering Flaw Acceptance Criteria for Salem Unit 1 Reactor Vessel Internals MRP-227-A Primary and Expansion Components, April 2017.

4.20 Westinghouse Drawing 647J081, Revision 8, 173-00-000-160 PWR Core Barrel – Lower, June 19, 1969.

4.21 Materials Reliability Program: Pressurized Water Reactor Internals Inspection and Evaluation Guidelines (MRP-227, Revision 1). EPRI, Palo Alto, CA: 2015. 3002005349.

4.22 Materials Reliability Program Letter, MRP 2018-003, Responses to NRC Requests for Additional Information for Electric Power Research Institute Topical Report MRP-227, Revision 1, Materials Reliability Program: Pressurized Water Reactor Internals Inspection and Evaluations Guideline (CAC No. M7740), January 30, 2018. (ML18038A875)

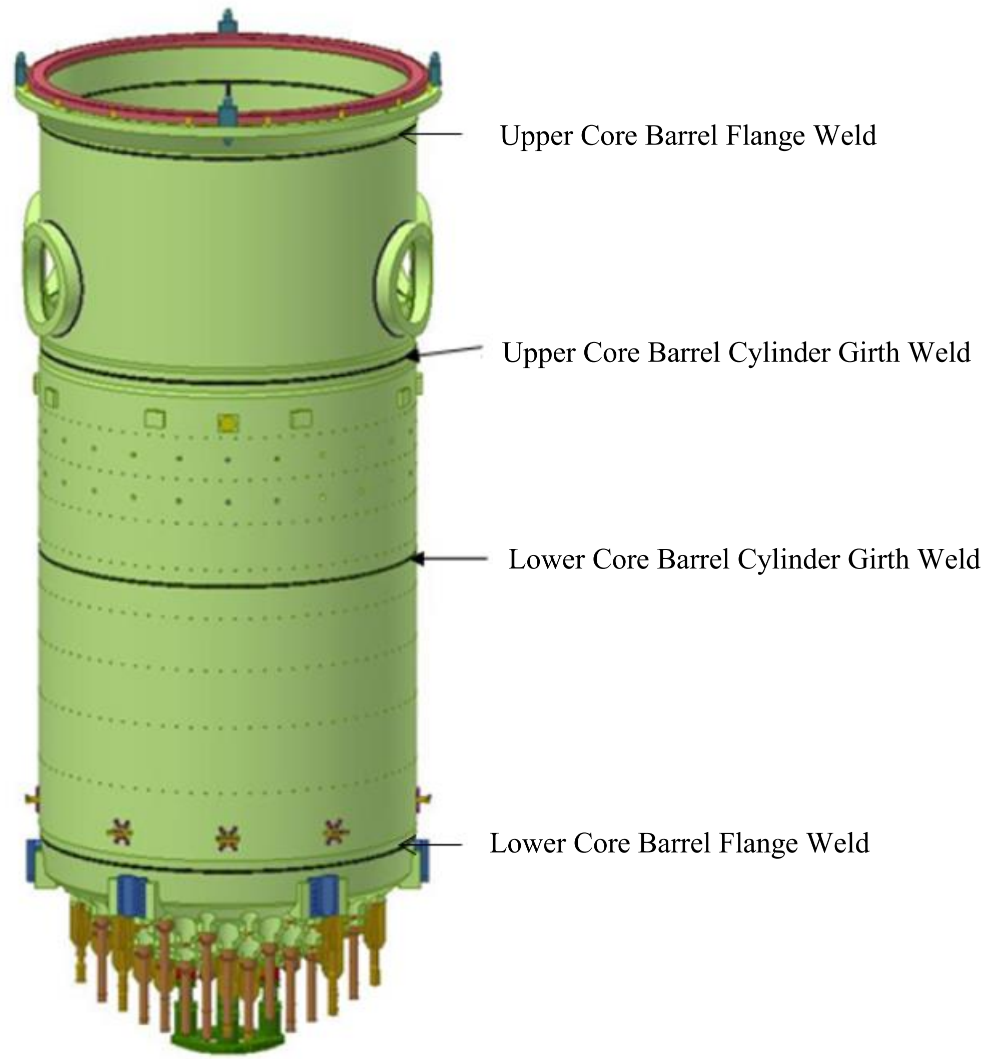
4.23 Materials Reliability Program Letter, MRP 2018-011, Transmit Supplemental Information Regarding EPRI Technical Report MRP-227-Revision 1, Materials Reliability Program: Pressurized Water Reactor Internals Inspection and Evaluation Guidelines, May 17, 2018.

4.24 Materials Reliability Program: Pressurized Water Reactor Issue Management Table, PWR-IMT Consequence of Failure (MRP-156). EPRI, Palo Alto, CA: 2005. 1012110.

5. Preparer/Reviewer/Approval

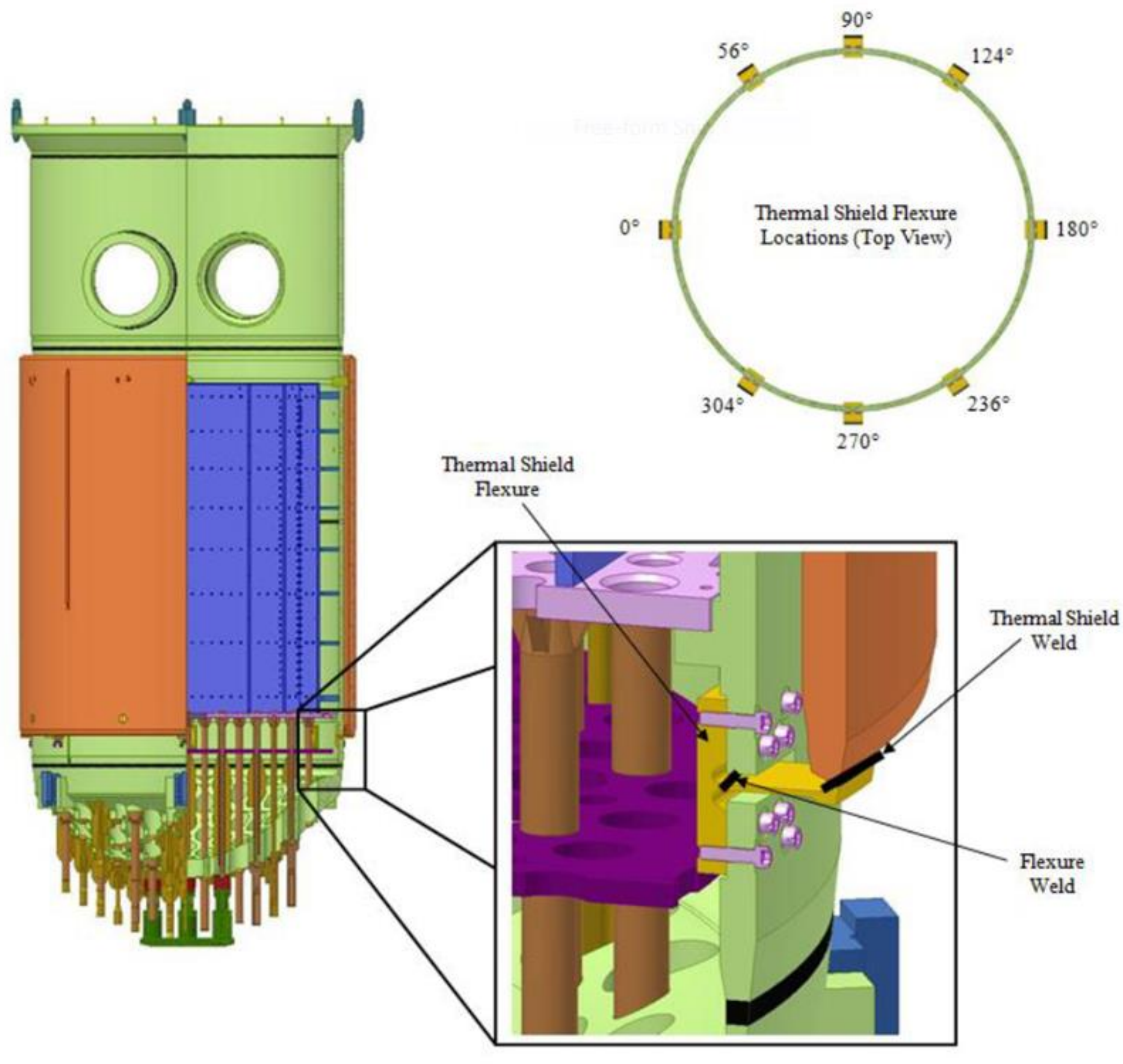
Preparer:	Patrick Fabian	[signature]	Date: 8-1-2018
Reviewer:	Tim Giles	SAP 70201108 op 10	Date: 8-1-2018
Independent Review:	Sam Speer	[signature]	Date: 8-1-18
Salem Site Approval:	Mike Ambrosino	M. Ambros	Date: 8-1-18
Corporate Approval:	Ali Fakhar	[signature]	Date: 8/1/2018

Attachment 1



Westinghouse Core Barrel and Welds

Attachment 2



Westinghouse Thermal Shield Flexures