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U. S. Nuclear Regulatory Commission
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Edwin I. Hatch Nuclear Plant – Units 1 and 2
10 CFR 50.55a Request for Alternative HNP-ISI-ALT-05-04
Implementation of BWRVIP Documents In Lieu of Certain B-N-1 and B-N-2 Examinations

Ladies and Gentlemen:

Pursuant to 10 CFR 50.55a(z)(1), Southern Nuclear Operating Company (SNC) requests Nuclear Regulatory Commission (NRC) approval of proposed Alternative HNP-ISI-ALT-05-04. This Alternative will allow alternative examination methods for certain American Society of Mechanical Engineers Section XI Class 1 components at the Edwin I. Hatch Nuclear Plant (HNP), Units 1 and 2.

SNC requests approval of the Alternative by January 15, 2019 to allow its use during the Unit 2 spring 2019 refueling outage.

This letter contains no NRC commitments. If you have any questions, please contact Jamie Coleman at 205.992.6611.

Respectfully submitted,

C. A. Gayheart
Regulatory Affairs Director

CAG/RMJ

Enclosures: 1. Request for Alternative HNP-ISI-ALT-05-04
2. NEI 03-08 Revision 3 Appendix C

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Enclosure 1

Request for Alternative HNP-ISI-ALT-05-04

1. ASME Code Component(s) Affected

Code Class:	ASME Section XI Code Class 1
Component Numbers:	None
Code References:	ASME Section XI, 2007 Edition with 2008 Addenda
Examination Category:	B-N-1, B-N-2
Item Number(s):	B13.10, B13.20, B13.30, and B13.40

2. Requested Approval Date

Approval is requested by January 15, 2019

3. Applicable ASME Code Requirements

ASME Section XI, 2007 Edition through the 2008 Addenda requires the examination of components within the Reactor Pressure Vessel (RPV). These examinations are included in Table IWB-2500-1, Examination Categories B-N-1 and B-N-2 and identified with the following Item Numbers:

B13.10 Examine accessible areas of the reactor vessel interior each period by the VT-3, visual examination method (B-N-1); includes only those spaces above and below the core made accessible by removal of components during normal refueling outages.

B13.20 Examine accessible interior welded attachments within the beltline region each interval by the VT-1, visual examination method (B-N-2).

B13.30 Examine accessible interior welded attachments beyond the beltline region each interval by the VT-3, visual examination method (B-N-2).

B13.40 Examine the accessible surfaces of welded core support structures each interval by the VT-3, visual examination method (B-N-2).

These examinations are performed to assess the structural integrity of the reactor vessel interior, its welded attachments, and the welded core support structure within the boiling water reactor pressure vessel.

4. Reason for Request

In accordance with 10 CFR 50.55a(z)(1), Hatch Nuclear Plant Units 1 and 2 (HNP) is requesting NRC approval of a proposed alternative to the Code requirements identified above on the basis that the use of the Boiling Water Reactor Vessel Internals Project (BWRVIP) guidelines discussed below provide an acceptable level of quality and safety.

The BWRVIP Inspection and Evaluation (I&E) Guidelines recommend specific inspection by BWR owners to identify material degradation with BWR components. A wealth of inspection data has been gathered during these inspections across the BWR fleet. The BWRVIP I&E Guidelines focus on specific and susceptible components, specify appropriate inspection methods capable of identifying known or potential degradation mechanisms, and require re-examination at appropriate intervals. The scope of the I&E Guidelines exceeds that of ASME Section XI and, in most instances, include components that are not part of the ASME Section XI jurisdiction.

Use of this proposed alternative will maintain an adequate level of quality and safety and avoid duplicate or unnecessary inspections, while conserving radiological dose.

5. Proposed Alternative and Basis for Use

Proposed Alternative

HNP requests authorization to utilize the alternative requirements of the BWRVIP I&E Guidelines in lieu of the requirements of ASME Code Section XI (including the examination method, examination volume, frequency, training, successive and additional examinations, flaw evaluations, and reporting). The proposed alternative includes:

For Examination Category B-N-1:

As an alternative to meeting ASME Section XI and performing a VT-3 examination of the RPV interior above and below the core made accessible by a normal refueling outage, HNP will implement the BWRVIP Guidelines listed below and as outlined in Table 1, or the latest BWRVIP approved revision consistent with NEI 03-08 Revision 3 guidance (Enclosure 2 to this letter), to those documents listed herein.

- BWRVIP-03, "Reactor Pressure Vessel and Internals Examinations Guidelines"
- BWRVIP-18, Revision 2-A, "BWR Core Spray Internals Inspection and Flaw Evaluation Guidelines"
- BWRVIP-25, "BWR Core Plate Inspection and Flaw Evaluation Guidelines"
- BWRVIP-26-A, "BWR Top Guide Inspection and Flaw Evaluation Guidelines"
- BWRVIP-41, Revision 4, "BWR Jet Pump Assembly Inspection and Evaluation Guidelines"
- BWRVIP-47-A, "BWR Lower Plenum Inspection and Evaluation Guidelines"

For Examination Category B-N-2:

As an alternative to meeting ASME Section XI and performing a VT-1 or VT-3, as required by ASME Section XI, examination of the RPV welded attachments and welded core support structures, HNP will take credit for implementation of the guidelines below

and as outlined in Table 1, or the latest BWRVIP approved revision consistent with NEI 03-08 Revision 3, to those documents listed herein.

- BWRVIP-03, "Reactor Pressure Vessel and Internal Examinations Guidelines"
- BWRVIP-38, "BWR Shroud Support Inspection and Flaw Evaluation Guidelines"
- BWRVIP-48-A, "Vessel ID Attachment Weld Inspection and Flaw Evaluation Guidelines"
- BWRVIP-76, Revision 1-A, "BWR Core Shroud Inspection and Flaw Evaluation Guidelines"

When a BWRVIP I&E Guideline refers to ASME Section XI, the technical requirements of ASME Section XI as described by the BWRVIP Guideline will be met, but the examination and evaluation are implemented through the BWRVIP program as defined by BWRVIP-94, "BWRVIP Vessel and Internals Project Program Implementation Guide".

The HNP reactor vessel internals inspection programs have been developed and implemented to satisfy the requirements of BWRVIP-94. It is recognized that the BWRVIP executive committee periodically revises the BWRVIP guidelines to address industry operating experience, include enhancements to inspection techniques, and add or adjust flaw evaluation methodologies. BWRVIP-94, Revision 2 states that where guidance in existing BWRVIP documents has been supplemented or revised by subsequent correspondence approved by the BWRVIP Executive Committee, the vessel and internals program shall be modified to reflect the new requirements and implement the guidance within two refueling outages, unless a different schedule is specified by the BWRVIP.

If new guidance approved by the Executive Committee includes changes to NRC approved BWRVIP guidance, the guidance shall be implemented in accordance with NEI 03-08, Revision 3, or after NRC approves the changes, which generally means publication of a "-A" document or equivalent. Table 1 compares the identified BWRVIP requirements to the corresponding ASME Section XI requirements.

Any deviations from the referenced BWRVIP Guidelines for the duration of the proposed alternative will be appropriately documented and communicated to the NRC, per the BWRVIP Deviation Disposition Process.

For conditions that require successive examinations in accordance with IWB-2420 that are identified before NRC approval of this request, the successive examinations will be completed to meet ASME Section XI. For conditions identified after NRC approval, successive examinations will be performed in accordance with the applicable BWRVIP Guideline and applicable evaluations.

In the event that conditions are identified that require repair or replacement and the component is within the jurisdiction of ASME Section XI (i.e., welded attachments to the

RPV or welded Core Support Structure), the repair or replacement activities will be performed in accordance with ASME Section XI, Article IWA-4000. Preservice examinations after repair or replacement activities and subsequent examinations will be in accordance with the applicable BWRVIP Guideline.

Basis for Use

As part of the BWRVIP initiative, the BWR reactor internals and attachments were subjected to a safety assessment to identify those components that provide a safety function and to determine if long-term actions were necessary to ensure continued safe operation. The safety functions considered are those associated with: (1) maintaining a coolable geometry, (2) maintaining control rod insertion times, (3) maintaining reactivity control, (4) assuring core cooling and (5) assuring instrumentation availability. The results of the safety assessment are documented in BWRVIP-06 Revision 1-A "BWR Vessel and Internals Project Safety Assessment of BWR Internals." As a result of BWRVIP-06 Revision 1-A, component specific BWRVIP guidelines were developed providing appropriate examination and evaluation requirements to address the specific component safety function and potential degradation mechanism.

Along with the component specific guidelines, the BWRVIP has established a reporting protocol for examination results and deviations. The NRC has agreed with the BWRVIP approach in principal and has issued Safety Evaluations for many of these guidelines (see References).

As additional justification, Attachment 1, "Comparison of Code Examination Requirements to BWRVIP Examination Requirements," provides specific examples which compare the inspection requirements of ASME Code Section XI Table IWB-2500-1, Category B-N-1 and B-N-2, Item Numbers B13.10, B13.20, B13.30 and B13.40 to the inspection requirements in the BWRVIP documents. This comparison also includes a discussion of the inspection methods.

Therefore, use of the BWRVIP guidelines as an alternative to ASME Section XI, as shown by the comparison provides an acceptable level of quality and safety.

6. Duration of Proposed Alternative

This proposed alternative will be used for the Fifth Ten-Year Interval of the Inservice Inspection Program for Hatch Nuclear Plants Units 1 and 2.

7. Precedents

Similar Request for Alternatives have been previously approved for the following licensees.

A. US NRC Letter to FirstEnergy Nuclear Operating Company, "Perry Nuclear Power Plant, Unit No. 1 – Approval of Alternative to Use BWRVIP Guidelines in lieu of Certain ASME

Code Requirements (CAC NO. MG0149; EPID 2017-LLR-0112)(L-17-183), dated January 29, 2018 (Accession Number ML18023A625).

B. US NRC Letter to Exelon Generation Company, LLC, "LaSalle County Station, Units 1 and 2, Relief from the Requirements of the ASME Code and OM Code RE: Relief Requests I4R-02, I4R-03, I4R-06, I4R-07, and I4R-09, Proposed Alternatives to Various Inservice Inspection Interval (ISI) Requirements of the American Society of Mechanical Engineers (ASME Code), Section XI, 2007 Edition with the 2008 Addenda for the Fourth 10-Year Interval (EPID NOS. L-2017-LLR-0038 (CAC NOS. MF9760 and MF9761), L-LR-2017-0076 (CAC NOS. MF9762 and MF9763), L-2017-LLR-0033 (CAC NOS. MF9766 and MF9767), L-2017-LLR-0035 (CAC NOS. MF9770 and MF9771), and L-2017-LLR-0037(CAC NOS. MF9768 and MF9769)), dated November 17, 2017, (Accession Number ML17305B279).

C. US NRC Letter to Exelon Nuclear Generation Company, LLC, "Safety Evaluation of Relief Requests I4R-02 and I4R-10 for the Fourth 10-Year Interval of the Inservice Inspection Program for Limerick Generating Station, Units 1 and 2 (CAC NOS. MF7587 and MF7588), dated November 21, 2016 (Accession Number ML16301A401).

D. US NRC Letter to Entergy Nuclear Operations, "Grand Gulf Nuclear Station, Unit 1 – Request for Relief GG-ISI-017, Alternative to use Boiling Water Reactor Vessel and Internals Project Guidelines in lieu of specific ASME Code Requirements (TAC No. MF2357)," dated June 30, 2014 (Accession Number ML14148A262).

E. US NRC Letter to Entergy Nuclear Operations, "River Bend, Unit1 – Request for Relief No. RBS-ISI-019, Alternative to use Boiling Water Reactor Vessel and Internals Project Guidelines in lieu of ASME Code, Section XI Requirements for the Fourth 10-Year Inservice Inspection Interval (TAC NO. MG1867), dated May 30, 2014 (Accession Number ML14127A327).

F. US NRC Letter to Exelon Generation Company, LLC, "Dresden Nuclear Power Station, Units 2 and 3 – Safety Evaluation in support of Request for Relief associated with the Fifth 10-Year Inservice Inspection Interval Program (TAC NOS. ME9682, ME9684, ME9685, ME9686, ME9687, ME9688, ME9689, ME9690, ME9691, ME9692, ME9693, ME9694, ME9695, ME9696, and ME9697), dated September 30, 2013 (Accession Number ML13260A585).

G. US NRC Letter to Exelon Generation Company, LLC, "Quad Cities Nuclear Power Station Units 1 and 2 – Safety Evaluation in support of Request for Relief associated with the Fifth 10 Year Interval Inservice Inspection Program (TAC NOS. ME9668, ME9669, ME9670, ME9671, ME9672, ME9674, ME9675, ME9676, ME9677, ME9678, ME9679, ME9680, ME9681), dated September 30, 2013 (Accession Number ML13267A097).

H. US NRC Letter to Exelon Nuclear, "Oyster Creek Nuclear Generating Station – Relief from the Requirements of the ASME Code, Relief Request No. I5R-01 (TAC NO. ME9490), dated August 5, 2013 (Accession Number ML13169A062).

8. References

A. Letter from K. Hsueh (NRC) to BWRVIP, "U.S. Nuclear Regulatory Commission Approval Letter for Electric Power Research Institute Topical Report, BWRVIP-18, Revision 2-A, BWR [Boiling Water Reactor] Vessel and Internals Project, BWR Core Spray Internals Inspection and Flaw Evaluation Guidelines (TAC No. MF8415)," dated December 21, 2016.
B. US NRC Letter to BWRVIP, dated December 19, 1999, "Final Safety Evaluation of BWRVIP Vessel and Internals Project, "BWR Vessel and Internals Project, BWR Core Plate Inspection and Flaw Evaluation Guidelines (BWRVIP-25)," EPRI Report TR-107284, December 1996 (TAC NO. M97802).
C. US NRC Letter to BWRVIP, dated September 9, 2005, "NRC Approval Letter of BWRVIP-26-A, "BWR Vessel and Internals Project Boiling Water Reactor Top Guide Inspection and Flaw Evaluation Guidelines."
D. US NRC Letter to BWRVIP, dated July 24, 2000, "Final Safety Evaluation of the "BWR Vessel and Internals Project, BWR Shroud Support Inspection and Flaw Evaluation Guidelines (BWRVIP-38)," EPRI Report TR-108823 (TAC NO. M99638).
E. US NRC Letter to BWRVIP, dated September 9, 2005, "NRC Approval Letter of BWRVIP-47-A, "BWR Vessel and Internals Project Boiling Water Reactor Lower Plenum Inspection and Flaw Evaluation Guidelines."
F. US NRC Letter to BWRVIP, dated July 25, 2005, "NRC Approval Letter of BWRVIP-48-A, "BWR Vessel and Internals Project Vessel ID Attachment Weld Inspection and Flaw Evaluation Guidelines."
G. BWRVIP-76NP, Revision 1: "BWR Vessel and Internals Project BWR Core Shroud Inspection and Flaw Evaluation Guidelines," dated May 2011 (ML11195A182).
H. Letter from Chairman, BWR Vessel and Internals Project to NRC, "Project No. 704 – BWRVIP Program Implementation Guide (BWRVIP-94NP, Revision 2)," dated September 22, 2011 (ML11271A058).

Table 1								
ASME Item No. Table IWB-2500-1	Component	ASME Exam Scope	ASME Exam Type	ASME Frequency	Applicable BWRVIP Document	BWRVIP Exam Scope	BWRVIP Exam Type	BWRVIP Frequency
B13.10	Reactor Vessel Interior	Spaces above and below the reactor core made accessible for examination by removal of components during a normal refuel outages.	VT-3	Each Period	None	While there is not a specific BWRVIP Guideline[1] that addresses the scope of B-N-1, the examinations performed by BWRVIP-18-R2-A, BWRVIP-25, BWRVIP-26-A, BWRVIP-41-R4, and BWRVIP-47-A, provide a general overview of the reactor interior which meets the intent of B-N-1 inspection requirements.		
B13.20	Interior Attachments within Beltline - Riser Braces	Accessible Welds	VT-1	Each 10-year Interval	BWRVIP-48-A Table 3-2	Riser Brace Attachment	EVT-1	100% in first 12 years, 25% during each subsequent 6 years
	Lower Surveillance Specimen Holder Brackets				BWRVIP-48-A, Table 3-2	Bracket Attachment	VT-1	Each 10-Year Interval
B13.30	Interior Attachments beyond Beltline - Steam Dryer Hold-down Brackets	Accessible Welds	VT-3	Each 10-year interval	BWRVIP-48-A, Table 3-2	Bracket Attachment	VT-3	Each 10-Year Interval
	Guide Rod Brackets				BWRVIP-48-A, Table 3-2	Bracket Attachment	VT-3	Each 10-Year Interval
	Steam Dryer Support Brackets				BWRVIP-48-A, Table 3-2	Bracket Attachment	EVT-1	Each 10-Year Interval

[1] For the specific BWRVIP Guideline being used, refer to "Proposed Alternative and Bases for Use" Section

Table 1								
ASME Item No. Table IWB-2500-1	Component	ASME Exam Scope	ASME Exam Type	ASME Frequency	Applicable BWRVIP Document	BWRVIP Exam Scope	BWRVIP Exam Type	BWRVIP Frequency
	Feedwater Sparger Brackets				BWRVIP-48-A, Table 3-2	Bracket Attachment	EVT-1	Each 10-Year Interval
	Core Spray Piping Brackets				BWRVIP-48-A, Table 3-2	Bracket Attachment	EVT-1	Every 4 Refueling Cycles
	Upper Surveillance Specimen Holder Brackets				BWRVIP-48-A, Table 3-2	Bracket Attachment	VT-3	Each 10-Year Interval
	Shroud Support (Weld H9) including gussets				BWRVIP-38, 3.1.3.2, Figures 3-2 and 3-5	Weld H-9 including gussets	EVT-1 or UT	Maximum of 6 years for EVT-1, Maximum of 10 years for UT Note: Hatch Unit 2 configuration does not have gussets.
B13.40	Integrally Welded Core Support Structure	Accessible Surfaces	VT-3	Each 10-year interval	BWRVIP-38, 3.1.3.2, Figures 3-2 and 3-5	Shroud support welds H8 and H9 including gussets	EVT-1 or UT	Based on as-found conditions, to a maximum 6 years for EVT-1, 10 years for UT where accessible
	Shroud Horizontal Welds				BWRVIP-76 R1-A, 2.2	Welds H1-H7 as applicable	UT or EVT-1	Note: Hatch has installed Tie Rods on both units and will perform the required examinations on the tie rods in lieu

Table 1								
ASME Item No. Table IWB-2500-1	Component	ASME Exam Scope	ASME Exam Type	ASME Frequency	Applicable BWRVIP Document	BWRVIP Exam Scope	BWRVIP Exam Type	BWRVIP Frequency
								of examining the Shroud Horizontal Welds.
	Shroud Vertical Welds				BWRVIP-76 R1-A, 2.3	Vertical Welds as applicable	EVT-1 or UT	Maximum 10 years for UT or EVT-1 based on inspection of horizontal welds, until inspection intervals are established for each weld. Note: Hatch has established inspection intervals for all vertical welds, with a maximum 10 year interval

ATTACHMENT 1
Comparison of ASME Code Section XI Examination
Requirements to BWRVIP Examination Requirements

The following provides a comparison of the examination requirements provided in ASME Code Section XI Table IWB-2500-1, Examination Category B-N-1 and B-N-2, Item Numbers B13.10, B13.20, B13.30, and B13.40, to the examination requirements in the BWRVIP Guidelines. Specific BWRVIP Guidelines are provided as examples for comparisons. This comparison also includes a discussion of the examination methods.

1. **Code Requirement – B13.10 – Reactor Vessel Interior Accessible Areas (B-N-1)**

   The ASME Section XI Code requires a VT-3 examination of reactor vessel accessible areas, which are defined as the spaces above and below the core made accessible during normal refueling outages. The frequency of these examinations is specified as the first refueling outage, and at intervals of approximately 3 years during the first inspection interval, and each period during each successive 10-year Inspection Interval. Typically, these examinations are performed every other refueling outage of the Inspection Interval. This examination requirement is a non-specific requirement that is a departure from the traditional Section XI examinations of welds and surfaces. As such, this requirement has been interpreted and satisfied differently among different utilities and inspection services vendors across the nuclear fleet. Based on the acceptance criteria specified in IWB-3520.2, the examination is to identify relevant conditions such as distortion or displacement of parts, loose, missing, or fractured fasteners, foreign material, corrosion, erosion, or accumulation of corrosion products, wear, and structural degradation.

   Portions of the various examinations required by the applicable BWRVIP Guidelines require regular access to accessible areas of the reactor vessel during refueling outage. Examination of Core Spray Piping and Spargers (BWRVIP-18-R2A), Core Plate Inspection (BWRVIP-25-A), Top Guide (BWRVIP-26-A), Jet Pump Welds and Components (BWRVIP-41-R4), and Lower Plenum Components (BWRVIP-47-A), provides such access. Locating and examining specific welds and components within the reactor vessel areas above, below (if accessible), and surrounding the core (annulus area) entails access by remote camera systems that essentially perform equivalent VT-3 examination of these areas or spaces as the specific weld or component examinations are performed. This provides an equivalent method of visual examination on a more frequent basis than that required by the ASME Section XI Code. Evidence of wear, structural degradation, loose, missing, or displaced parts, foreign materials, and corrosion product buildup can be, and has been observed during the course of implementing these BWRVIP examination requirements.

   Therefore, the requirements specified by the BWRVIP Guidelines meet or exceed the subject Code requirements for examination method and frequency of the interior of the

reactor vessel. Accordingly, these BWRVIP examination requirements provide an acceptable level of quality and safety as compared to the subject Code requirements.

2. Code Requirement – B13.20 – Interior Attachments Within the Beltline (B-N-2)

The ASME Section XI Code requires a VT-1 examination of accessible reactor interior surface attachment welds within the beltline each 10-year interval. For HNP Units 1 and 2, this includes the jet pump riser brace weld-to-vessel wall and the lower surveillance specimen support bracket welds-to-vessel wall. In comparison, the BWRVIP requires the same examination method and frequency for the lower surveillance specimen support bracket welds, and requires an EVT-1 examination on the remaining attachment welds in the beltline region in the first 12 years, and then 25% during each subsequent 6 years.

The jet pump riser brace examination requirements are provided below to show a comparison between the Code and the BWRVIP examination requirements.

<u>Comparison to BWRVIP Requirements – Jet Pump Riser Braces (BWRVIP-48-A)</u>

- The ASME Code requires a 100% VT-1 examination of the jet pump riser brace-to-reactor vessel wall pad welds each 10-year Interval.
- The BWRVIP requires an EVT-1 baseline examination of 100% of the jet pump riser brace-to-reactor vessel wall pad welds in the first 12 years with at least 50% being inspected in the first 6 years. Reinspection consists of 25% during each subsequent 6-year period.
- BWRVIP-48-A specifically defines the susceptible regions of the attachment that are to be examined.

The Code VT-1 examination is conducted to detect discontinuities and imperfections on the surfaces of components, including such conditions as cracks, wear, corrosion, or erosion. The BWRVIP enhanced VT-1 (EVT-1) is conducted to detect discontinuities and imperfections on the surface of components and is additionally specified to detect potentially very tight cracks characteristic of fatigue and intergranular stress corrosion cracking (IGSCC), the relevant degradation mechanisms for these components. General wear, corrosion, or erosion although generally not a concern for inherently tough, corrosion resistant stainless steel material, would also be detected during the process of performing a BWRVIP EVT-1 examination.

The ASME Code visual examination method requires (depending on applicable ASME Edition) that a letter character with a height of 0.044 inches can be read. The BWRVIP EVT-1 visual examination method requires the same 0.044-inch resolution on the examination surface and additionally, a more stringent viewing angle for remote video examination, the performance of a cleaning assessment and cleaning as necessary. While the jet pump riser brace configuration varies depending on the vessel

manufacturer, BWRVIP-48-A includes diagrams for each configuration and prescribes examination for each configuration.

The resolution standards used for BWRVIP EVT-1 examinations utilize the same Code characters, thus assuring at least equivalent resolution compared to the Code. Although the BWRVIP examination may be less frequent, it is a more comprehensive method. Therefore, the BWRVIP guidance provides an acceptable level of quality and safety to that provided by the ASME Code.

3. Code Requirement – B13.30 – Interior Attachments Beyond the Beltline Region (B-N-2)

The ASME Section XI Code requires a VT-3 examination of accessible reactor interior surface attachment welds beyond the beltline each 10-year Interval. For HNP Units 1 and 2, this includes the core spray piping primary, the upper surveillance specimen support bracket welds-to-vessel wall, the feedwater sparger support bracket welds-to-reactor vessel wall, the steam dryer support bracket welds-to-reactor vessel wall, the guide rod support bracket weld-to-reactor vessel wall, the shroud support plate-to-vessel welds, and shroud support gussets. BWRVIP-48-A requires as a minimum the same VT-3 examination method as the Code for some of the interior attachment welds beyond the beltline region, and in some cases specifies an enhanced visual examination technique EVT-1 for these welds. For those interior attachment welds that have the same VT-3 method of examination, the same scope of examination (accessible welds), the same examination frequency (each 10-year interval) and ASME Section XI flaw evaluation criteria, the level of quality and safety provided by the BWRVIP requirements are equivalent to that provided by the ASME Code.

The core spray piping bracket-to-vessel attachment weld is used as an example for comparison between the Code and BWRVIP examination requirements as discussed below:

<u>Comparison to BWRVIP Requirements – Core Spray Piping Bracket Welds (BWRVIP-48-A)</u>

- The Code examination requirement is a VT-3 examination of each weld every 10 years.
- The BWRVIP examination requirement is an EVT-1 for the core spray piping bracket attachment welds with each weld examined every four cycles (8 years for units with a 2-year fuel cycle)

The BWRVIP examination method EVT-1 has superior flaw detection and sizing capability than the Code VT-3, the examination frequency is greater than the Code requirements, and the same flaw evaluation criteria are used.

The Code VT-3 examination is conducted to detect component structural integrity by ensuring the components general condition is acceptable. An enhanced EVT-1 is conducted to detect discontinuities and imperfections on the examination surfaces, including such conditions as tight cracks caused by IGSCC or fatigue, the relevant degradation mechanisms for BWR internal attachments. Additionally, BWRVIP-48 guidance requires indications detected by an EVT-1 to be examined by ultrasonic testing to determine if the indication has propagated into the reactor vessel base material.

Therefore, with the EVT-1 examination method, the same examination scope (accessible welds), an increased examination frequency (8 years instead of 10 years) in some cases, and the same flaw evaluation criteria (ASME Code Section XI), the level of quality and safety provided by the BWRVIP criteria is superior to that provided by the ASME Code.

4. Code Requirement – B13.40 – Core Support Structure (B-N-2)

The ASME Code requires a VT-3 examination of accessible surfaces of the welded core support structure each 10-year interval. For HNP Units 1 and 2, the welded core support structure has primarily been considered the shroud support structure, including the shroud support plate (annulus floor), the shroud support ring, the shroud support welds, and the shroud support gussets. In later designs, the shroud itself is considered part of the welded core support structure. Historically, this requirement has been interpreted and satisfied differently across the industry. The proposed alternate examination replaces this ASME requirement with specific BWRVIP guidelines that examine susceptible locations for known relevant degradation mechanisms.

For integrally welded core support structure components, the BWRVIP requires an EVT-1 or UT of core support structures. The core shroud is used as an example for comparison between the Code and BWRVIP examination requirements as shown below.
The BWRVIP recommended examinations specify locations that are known to be vulnerable to BWR relevant degradation mechanisms rather than "all accessible surfaces". The BWRVIP examination methods (EVT-1 or UT) are superior to the Code required VT-3 for flaw detection and characterization.

<u>Comparison to BWRVIP Requirements - BWR Core Shroud Examination and Flaw Evaluation Guideline (BWRVIP-76 R1-A)</u>

- ASME Section XI requires a VT-3 examination of accessible surfaces each 10-year interval.
- For repaired shrouds, the BWRVIP requires either an EVT-1 examination from the inside and outside surface where accessible, or an ultrasonic examination of welds which have not been structurally replaced with a shroud repair. Inspection frequency is at a calculated "end of interval" (EOI) that will vary depending upon the amount of flaws present, but not to exceed ten years.

BWRVIP recommended examinations of integrally welded core support structures are focused on the known susceptible areas of this structure, including the welds and associated weld heat affected zones. In many locations, the BWRVIP guidelines require a volumetric examination of the susceptible welds at a frequency identical to the Code requirement.

The BWRVIP examination frequency is equivalent to or more frequent than the examination frequency required by the Code. The superior flaw detection and characterization capability, with an equivalent or more frequent examination frequency and the comparable flaw evaluation criteria, results in the BWRVIP criteria providing a level of quality and safety equivalent to or superior to that provided by the Code requirements.

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Enclosure 2

NEI 03-08 Revision 3 Appendix C

APPENDIX C
DOCUMENT SCREENING

1 PURPOSE

This appendix defines a screening process that may be applied by any NEI 03-08 Issue Program (IP) to determine if a new or revised work product containing aging management guidance may be generically released for implementation by IP member utilities.

2 BACKGROUND

As a means of ensuring continued safe operation of reactor plants, NRC and industry have often agreed to use the topical report submittal and approval process to address materials degradation issues in lieu of regulatory action. However, as industry continues to progress the overall state of knowledge regarding materials degradation issues relevant to light-water reactor operation, there is an increasing need to revise or replace prior guidance, some of which has received prior NRC approval via safety evaluation (SE). Since implementation of NEI 03-08, industry generally has not implemented revised or replacement guidance that is less conservative in some way than previously approved guidance without NRC approval of the guidance changes. Although beneficial in assuring that aging management guidance changes are reasonable and technically sound, use of the topical report review and approval process is often an inefficient use of limited resources, requiring both industry and NRC to expend significant effort on topical report modifications having limited or no potential for a significant adverse impact on the capability of the aging management guidance within the topical report to provide reasonable assurance of continued safe operation. The screening process contained in this appendix is intended to alleviate this issue by providing IPs with a method that may be used to determine when revised or replacement guidance may be implemented without NRC review and approval.

3 APPLICABILITY

This screening process is applicable to revised and new work products prepared by IPs identified under the NEI 03-08 Materials Initiative (those IPs listed in Appendix A of this initiative document) containing guidance that either directly or indirectly affects aging management.

This process is intended to be applied in the context of U.S. licensing and is directly applicable only to U.S. licensed reactors operated by EPRI IP member utilities.

4 DEFINITIONS

Applicability Evaluation:
Process for determining if screening is applicable to an NEI 03-08 IP product (described in Section 5.1).

Generic Release for Implementation:
A determination that a new or revised IP product can be generically released for implementation means that there are no generic limitations preventing implementation by IP member utilities. However, each site is responsible to review its site-specific licensing and design bases, license renewal commitments, and inservice inspection (ISI) program relief requests and alternatives to

ensure that there are no plant-specific limitations that would preclude immediate implementation of portions or all of the new or revised guidance in the product.

IP Controlled Aging Management Guidance:
Guidance that can either directly or indirectly affect aging management of a SSC. To be IP controlled, the aging management guidance must also be in addition to existing regulation or ASME Code requirements.

Screening:
Process for determining if aging management guidance contained in an NEI 03-08 IP work product may be generically released for implementation by member utilities without NRC approval.

SSC:
System, Structure, or Component

Product:
The term "product(s)" or "work product(s)" is used in this appendix to mean those documents issued by the IPs to their members prescribing requirements, recommendations, or guidelines.

5 DOCUMENT SCREENING PROCESS

Sections 5.1 and 5.2 describe the process to be used to determine when an IP may direct member utilities to generically implement new or revised aging management guidance contained in work products without NRC approval.

Section 5.1 provides guidance for determining if a screening evaluation is applicable to an IP work product. The decision steps provided in Section 5.1 are intentionally limited in complexity, relating primarily to the product's intended use and status. The evaluation described in Section 5.1 can generally be performed without a detailed understanding of plant design, component function, degradation phenomena relevant to reactor primary systems, materials-related operating experience, or the details of the aging management program elements recommended within the product.

Section 5.2 provides guidance for a screening evaluation based on the details of the aging management element changes recommended within the IP product. The evaluation described in Section 5.2 must be performed by an individual having a fundamental understanding of component function, relevant degradation modes, fleet operating experience, component capability to tolerate degradation, and the capability of the inspection methods prescribed in the guidance to detect and characterize relevant degradation.

5.1 Applicability Evaluation

Figure 1 provides a set of decision steps that may be used by an IP to determine if screening is required prior to generically releasing the product for implementation. Table 1 provides an amplification of the decision steps shown in Figure 1, along with additional relevant implementation instructions and notes.

In the case that decision steps (1a) through (1e) in Figure 1 all result in YES answers, detailed screening as described in Section 5.2 is needed if the product is implemented by IP member utilities without NRC approval via SE.

If one or more of the decision steps in Figure 1 results in a NO answer, the product may be generically released for implementation without NRC approval without a screening evaluation.

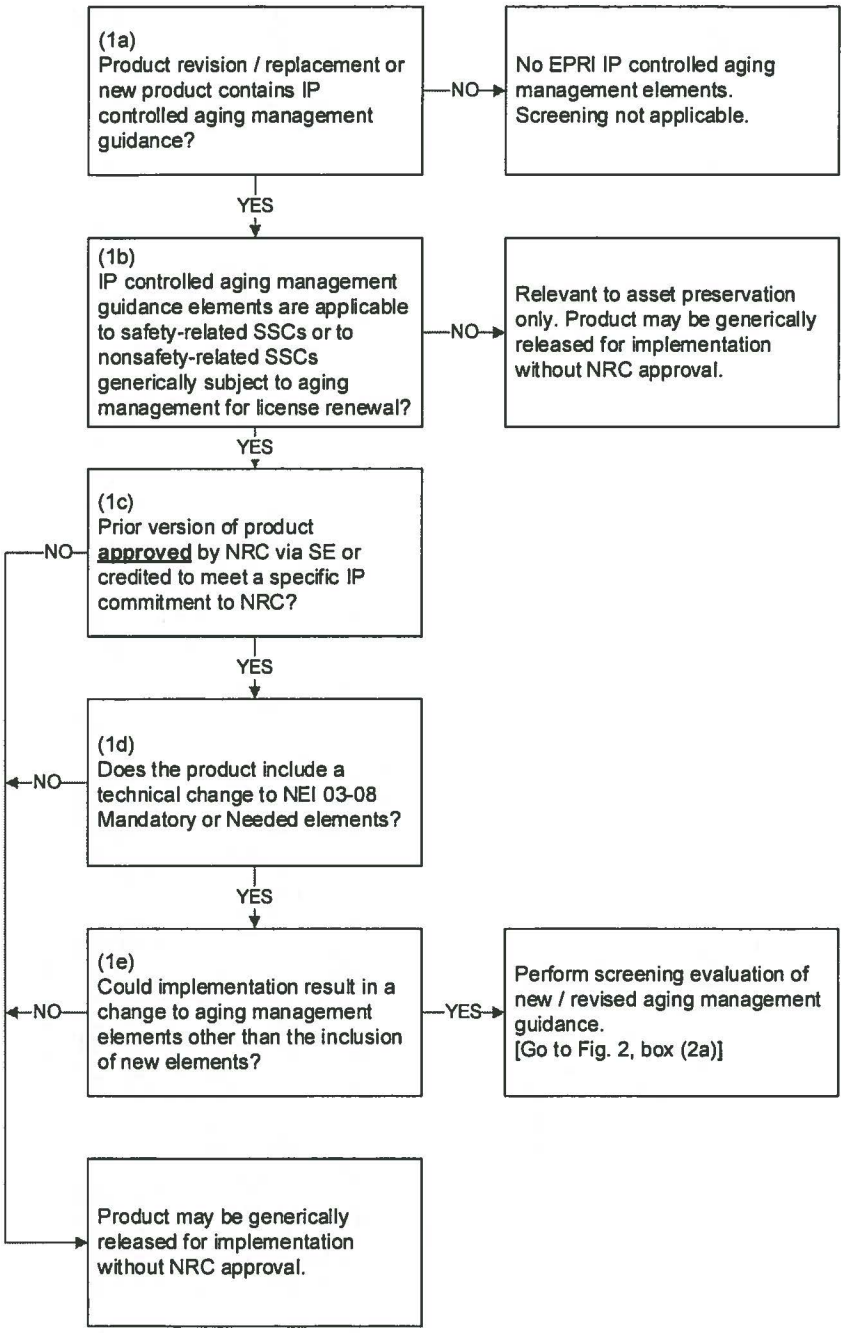

Figure 1: Applicability Evaluation Process

Table 1: Implementation Guidance for Use of Figure 1, Applicability Evaluation Process

Implementation Guidance	Technical Basis / Discussion
(1a) IP product revision / replacement or new product contains IP controlled aging management guidance?	The document screening process is applicable only when the IP product contains IP controlled aging management guidance – defined as guidance meeting the following conditions: 1) Guidance represents an augmentation of Regulatory or ASME Code requirements.[1] 2) Guidance can affect aging management activities, either directly or indirectly.[2] If the answer to (1a) is NO, then the product does not contain any IP controlled aging management elements. Submittal to NRC for approval via SE is not required. If the answer to (1a) is YES, proceed to question (1b).
(1b) IP controlled aging management guidance elements are applicable to safety-related SSCs or to nonsafety-related SSCs generically subject to aging management for license renewal?	Changes to aging management elements applicable only to SSCs not subject to aging management for license renewal cannot have an adverse impact on nuclear safety. Expanding the scope of components considered in this step to include SSCs generically subject to aging management for license renewal ensures that aging management guidance changes relevant to nonsafety-related SSCs whose failure could prevent satisfactory accomplishment of a safety-related function is conservatively evaluated.[3] If the answer to (1b) is NO, then product applicability is limited to asset preservation. Submittal to NRC for approval via SE is not required. If the answer to (1b) is YES, proceed to question (1c).

[1] If aging management implementation is ultimately controlled by regulation or by ASME Code, then the IP is not the governing organization and changes to aging management elements must be adopted outside the IP's NEI 03-08 implementation process.

[2] Direct aging management guidance elements include inspection method, scope, frequency, sample size, scope expansion requirements, supplemental examination requirements, evaluation methods and acceptance criteria. Indirect aging management guidance elements are those which support application of direct elements. Examples of indirect aging management elements include crack growth rate and fracture toughness correlations used for flaw evaluations and criteria for inspection relief related to mitigation status.

[3] A determination of components "generically" subject to aging management can be based on either NUREG-1801, Generic Aging Lessons Learned (GALL) Report or NUREG-2191, GALL Report for Second License Renewal.

(1c) Has ANY prior version of the product been approved by NRC via Safety Evaluation (SE) or is the product a direct replacement for guidance previously approved by NRC via SE?	In the case of multiple revisions to an IP product, this decision step is not limited to the immediately preceding revision. If ANY prior version of the product was approved by NRC via safety evaluation, answer this question "YES" and proceed to question (1d).[4] In the case of a new product, if the product directly replaces prior guidance that was approved by NRC via SE, answer this question "YES" and proceed to question (1d).
Does the product contain guidance credited to meet a specific IP commitment to NRC?	Although not common, in lieu of explicit NRC approval via SE, it is possible that an IP may commit to specific aging management guidance elements as part of interactions with NRC. Such aging management guidance elements should be treated similar to guidance approved by NRC via SE. Answer this question "YES" and proceed to question (1d). If the answers to these questions related to prior NRC SE and IP commitment to NRC are both NO, the product may be generically released for implementation.
(1d) Does the new or revised product include a technical change to NEI 03-08 Mandatory or Needed elements? [5]	Each IP is responsible for categorizing aging management elements as Mandatory, Needed, or Good Practice. Aging management elements that are deemed to be significant with regard to ensuring adequate management or to have risk significance are categorized by IPs as either Mandatory or Needed elements. If the answer to this question is YES, proceed to question (1e). If the answer to this question is NO, the revised or new product is not risk significant and may be generically released for implementation.

[4] In some cases, IP products containing aging management guidance that is more conservative than that approved by SE are implemented without submittal of the revised aging management guidance to NRC for approval via SE. This decision step ensures that all aging management guidance previously approved by NRC via SE is subjected to the significance decision steps provided in (1d) and (1e).

[5] See Section 3 of NEI 03-08, Revision 2.

(1e) Could implementation of the aging management guidance contained in the product result in a change to aging management elements other than the inclusion of new elements?	For any product revision that does not clearly result in equivalent or more conservative aging management guidance than that previously approved by NRC, screening evaluation in accordance with Section 5.2 must be performed if the product is not submitted to NRC for approval via SE. Go to Figure 2, evaluation step (2a). If the answer to this question is NO, NRC approval via SE is not required prior to generic release for implementation.[6]

[6] Although not required, an EPRI IP may still choose to submit a new or revised topical report for NRC review and approval.

5.2 Screening Evaluation

Figure 2 provides a process that should be applied to IP products determined to require screening based on the applicability evaluation performed consistent with Section 5.1. Table 2 provides an amplification of the decision steps shown in Figure 2, along with additional relevant implementation instructions and notes.

Within an IP work product, aging management element changes may be dispositioned independently if so desired. Further, if deemed appropriate by the IP, products may be generically released with instructions for partial implementation until such time as an NRC SE is received for any changes to aging management elements determined to require NRC approval prior to implementation. As such, the screening steps provided in Figure 2 and Table 2 are focused on aging management elements instead of IP products.

The screening evaluation steps described in this section should be performed by a qualified individual having a fundamental understanding of component function, relevant degradation modes, fleet operating experience, component capability to tolerate degradation, and the capability of the inspection methods prescribed in the guidance to detect and characterize relevant degradation.

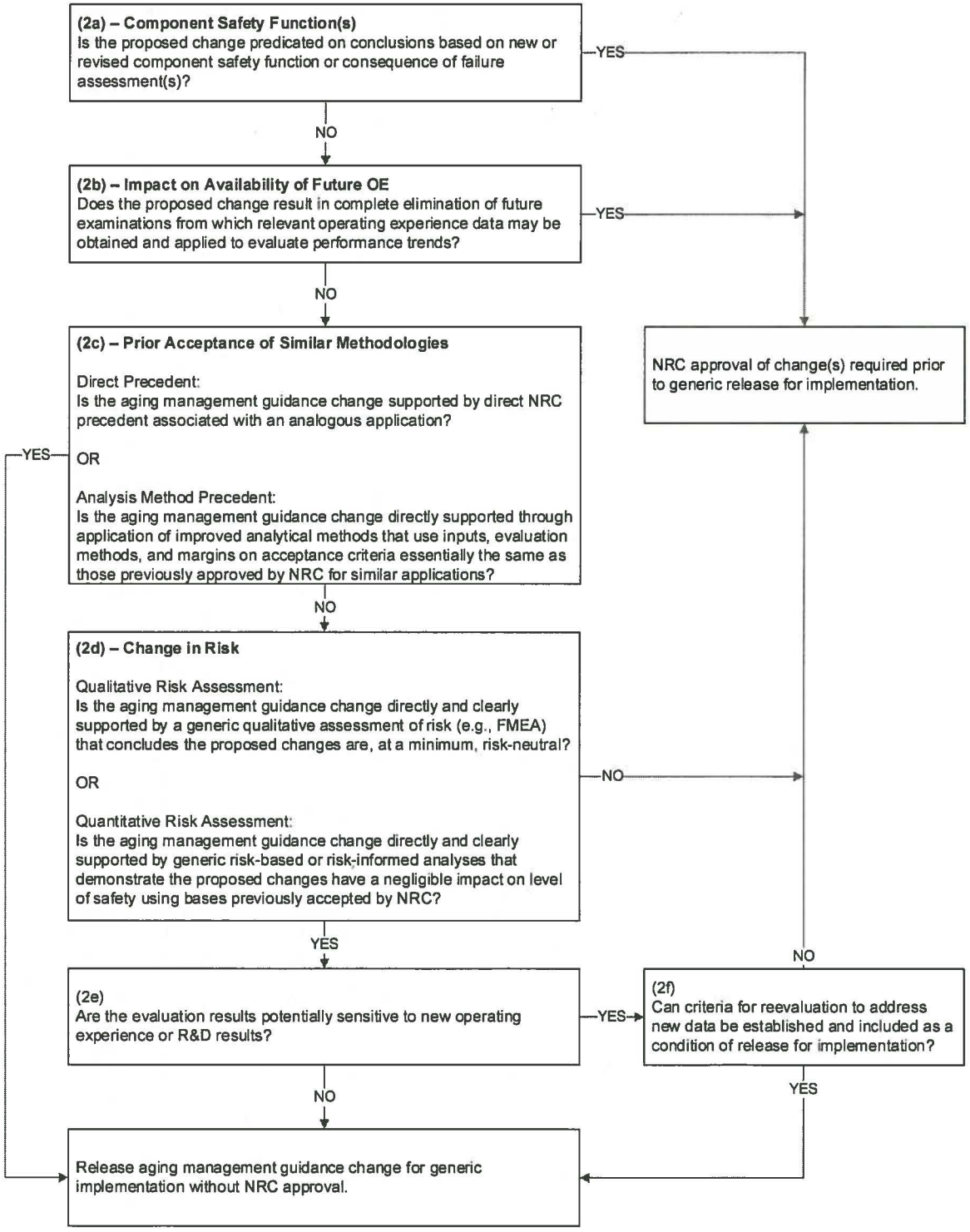

Figure 2: Screening Evaluation Process

Table 2: Implementation Guidance for Use of Figure 2, Screening Evaluation Process

Implementation Guidance	Technical Basis / Discussion
(2a) Component Safety Function Assessment	Changes to aging management elements that are predicated on new or revised assessments of component safety function or deterministic consequence of failure assessments are conservatively considered to represent changes that require NRC approval prior to generic release for implementation.
(2b) Impact on Availability of Future OE	A key feature of a robust aging management program is evaluation and appropriate incorporation of new knowledge related to materials degradation. Fleet inspection data is particularly valuable in assessing performance trends. Where component locations previously inspected by an aging management program are removed from future inspection scope without identification of reasonable surrogate locations remaining in the population of components inspected (whether in individual plants or within the fleet at large)[7], it is reasonable to obtain NRC approval prior to implementation.

[7] Appropriate surrogate locations may be fleet-based (i.e., surrogate locations need not be defined on a plant-specific basis).

(2c) Prior Acceptance of Similar Methodologies	Prior acceptance by NRC may be used in at least two ways: <u>Direct Precedent</u>: When prior approval from NRC via SE for an aging management method has been received for an essentially identical application, it is reasonable to conclude that an analogous approach applied to similar components for an essentially identical purpose need not be approved by NRC prior to implementation. <u>Analysis Method Precedent</u>: In cases, work performed after generation of aging management guidance has resulted in the development of improved analysis methods that have been accepted by NRC. These NRC approved analysis methods may be applied to additional component locations to refine the recommended aging management guidance. In this case, the precedent is set indirectly (i.e., based on analysis method) rather than directly (i.e., based on specific aging management elements). For determination based on analysis method precedent, the analysis application must be consistent with the purpose and intent of the precedent analysis application. Additionally, the analytical methods, key analysis assumptions and inputs, and acceptance criteria (including applied margins to minimum acceptable values) used must be essentially the same as those used in the analysis on which the precedence evaluation is based.

(2d) Risk Assessment	If a determination cannot be reached using step 2(c), then an assessment of risk may be applied to determine if the proposed aging management guidance significantly impacts overall level of safety. Assessments of risk may be either qualitative or quantitative in nature as described below. <u>Qualitative Risk Assessment</u>: Qualitative evaluation of field inspection data and R&D program has been extensively used by IPs as bases for development of aging management guidance.[8] Evaluation of new or improved data using a similar evaluation process may be used as a basis for modification of the recommended aging management elements. The method applied should ideally be consistent with methods previously applied and either directly or indirectly accepted by NRC in that the aging management guidance resulting from use of the method were approved by NRC.[9] <u>Quantitative Risk Assessment</u>: Quantitative risk-based methods may be used to demonstrate that the proposed changes to aging management elements do not represent a significant change in risk. For the purpose of this screening process, quantitative measures of risk may be defined in any of several ways, including but not limited to, core damage frequency and conditional probability of failure. Where applied, risk calculations should apply methods that have been either explicitly approved by NRC for similar use or that apply appropriate safety margins to account for differences in professional opinion regarding appropriate input assumptions and calculational procedures. Acceptance criteria must be consistent with those accepted by NRC for similar analytical evaluations. In all applications, the intent of any NRC conditions placed on the use and acceptability of similar analysis methods and resulting aging management elements must be considered in the determination.

[8] Failure modes and effects analysis (FMEA) is an example of a routinely applied qualitative risk assessment method. Results are based on categorization of component locations based on risk of degradation or on qualitative assessment of failure consequence rather than on calculation of probabilistic values (e.g., core damage frequency or conditional probability of failure).

[9] For example, MRP-227 applied a structured failure modes, effects, and criticality analysis (FMECA) process to evaluate PWR internals and determine appropriate aging management requirements. In approving MRP-227, it is established that NRC accepts as valid the FMECA process used by MRP. In the case where new data (either based on field OE or based on completed R&D) are used as inputs to a revised FMECA, the results are deemed not to require NRC approval prior to generic release for implementation so long as the FMECA is applied in a manner consistent with that used previously which has been accepted by NRC.

(2e)/(2f) Sensitivity to New Data	If a risk assessment consistent with item (2d) is used as a screening basis, it is recognized that the evaluation conclusions could be affected by new field OE or by R&D results. In the case that changes to aging management guidance are released for generic implementation without NRC approval and such changes could be sensitive to new data, it is reasonable that criteria be established for identifying any adverse trends in performance that could warrant adjustment of the applicable aging management guidance. Criteria for reevaluation must be established and managed appropriately by the responsible IP in a manner that ensures adverse performance trends are identified and addressed in a timely manner. This requirement is consistent with the approach used to maintain risk-informed ISI programs.

5.3 Review, Approval and Documentation

For IP products determined not to require NRC approval prior to generic release for implementation based on a determination using Section 5.1, formal review and approval of the determination is not required. The determination result is documented in the letter transmitting the IP product for committee review.

For IP products determined not to require NRC approval prior to release for generic implementation based on screening evaluation using Section 5.2, the details of the screening evaluation shall be documented and provided for committee review and approval in parallel with the product. Review of the screening evaluation is performed using the same committee-based consensus process applied to the product. Review and approval of the screening evaluation by IP member utilities occurs through application of the existing process under which products containing NEI 03-08 Mandatory or Needed elements require executive body approval before implementation.[10] Final documentation of the screening evaluation details including any analyses performed to support the evaluation (not just the determination result) should be documented either as an attachment to the IP letter transmitting the report to members for implementation or as an attachment or appendix to the IP product itself.

Finally, the introduction section of all IP products containing NEI 03-08 Mandatory or Needed elements should clearly state the report implementation status. Where detailed evaluation using Section 5.2 was used to determine that the product does not require NRC approval prior to release for implementation, the introduction section should also provide a reference to the screening evaluation so that program owners implementing the aging management guidance contained in the product can access the screening evaluation details if desired.

5.4 NRC Notification

Each IP will provide an annual information only notification to NRC regarding application of the screening process. The level of detail provided is left to the discretion of the IP. The annual information notification shall be reviewed by NEI. If areas of regulatory risk are identified in the review, the IP and NEI will work collaboratively to develop appropriate communication for the annual information only notification that minimizes the regulatory risks. However, as a minimum, the notification shall include a listing of work products that, during the annual reporting period, meet all of the following criteria:

1) Include IP controlled aging management guidance elements and,
2) Represent a revision of or replacement for a product previously approved by NRC via SE and,
3) Was evaluated by the screening process described in Section 5.2 and determined not to require NRC approval prior to release for generic implementation

Further, for each report listed based on these criteria, the notification shall also include a summary statement of the basis applied by the IP to determine that the product could be released for generic implementation without NRC approval. Evaluation details need not be

[10] Aging management guidance elements that are deemed important to assuring continued safe operation will be indicated as either Mandatory or Needed elements under the NEI 03-08 Materials Initiative.

provided within the annual notification. However, evaluation details shall be maintained by the responsible IP and made available to NRC upon formal request.

PLANT APPLICATION

The process steps described in Section 5 are intended to provide IP staff or qualified evaluators with an appropriate process to determine if an IP product can be generically released for implementation without NRC approval or if NRC approval via SE is needed prior to such a release. However, plants may identify limitations within the site-specific licensing basis, NRC commitments, or plant ISI program that conflict with immediate implementation of an IP product. These limitations must be resolved on a plant-specific basis. In no case should evaluations performed consistent with this appendix be considered to supersede or replace plant-specific limitations on aging management guidance implementation.