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UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE COMMISSION

In the Matter of	)	
	)	
ENTERGY NUCLEAR OPERATIONS, INC.	)	Docket No. 50-247-LR/50-286-LR
	)	
(Indian Point Nuclear Generating	)	
Units 2 and 3)	)	

NRC STAFF'S INITIAL STATEMENT OF POSITION ON
CONTENTION NYS-25 (REACTOR VESSEL INTERNALS)

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TABLE OF CONTENTS

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INTRODUCTION

In accordance with 10 C.F.R. § 2.1207(a) and the Atomic Safety and Licensing Board's ("Board") Orders in this proceeding,[1] the NRC Staff ("Staff") hereby submits its Statement of Position ("Statement") on the State of New York's ("New York" or "NYS") Contention NYS-25 (Reactor Vessel Internals). This Statement is supported by the prefiled written testimony of Dr. Allen Hiser, Jeffrey Poehler, and Gary Stevens ("Staff Testimony") (Exhibit ("Ex.") NRC000197), and the exhibits cited therein (including Exs. NRC000198 through NRC000229).[2]

Contention NYS-25 generally challenges the adequacy of the aging management program ("AMP") for reactor vessel internals ("RVI") at Indian Point Nuclear Generating Units 2 and 3 ("IP2" and "IP3"), to manage the effects of aging on RVI that may be subject to multiple aging effects through the period of extended operation. The Staff has carefully considered the assertions presented in this contention, as amplified in the testimony of New York's expert, Dr. Richard Lahey (Ex. NYS000296 and NYS000482), and the exhibits filed in support thereof. For

[1] *See* (1) "Order (Granting NRC Staff's Unopposed Time Extension Motion and Directing Filing of Status Updates)" (Feb. 16, 2012), at 1; (2) "Order (Clarification of Procedures for Evidentiary Filings)" (Oct. 18, 2011), at 2-3; (3) "Order (Procedures for Evidentiary Filings)" (Oct. 7, 2011), at 2; and (4) "Scheduling Order" (July 1, 2010), at 14.

[2] It should be noted that the Dr. Hiser and Mr. Poehler's testimony in support of NYS-25 is also being used to support the Staff's position with respect to RVI portion of NYS-38/RK-TC-5.

the reasons set forth below and in the testimony filed herewith, the Staff respectfully submits that Contention NYS-25 is lacking in merit and should be resolved in favor of Entergy Nuclear Operations, Inc. ("Entergy" or "Applicant") and the Staff.

BACKGROUND

Contention NYS-25 was filed by New York on November 30, 2007. As filed by New York, the contention stated as follows:

> NYS-25
>
> Entergy's license renewal application does not include an adequate plan to monitor and manage the effects of aging due to embrittlement of the reactor pressure vessels ("RPVs") and the associated internals.

"New York State Notice of Intention to Participate and Petition to Intervene" (Nov. 30, 2007) ("NY Petition"), at 223. In admitting the contention, the Board focused on Dr. Lahey's assertions regarding maintaining a coolable core geometry and new accident analysis for embrittled reactor vessel internal structures. *Entergy Nuclear Operations, Inc.* (Indian Point Units 2 and 3), 68 N.R.C. 43, 218 (July 31, 2008).

The bases for Contention NYS-25 were set forth in New York's Petition. As set forth therein, and as summarized by the Board in LBP-08-13, New York generally asserted that the RVI AMP for IP2 and IP3 is inadequate in that (1) the application, at that time, committed to implement RVI AMP as approved by the NRC; and that (2) Entergy has not considered the impact of design-basis accident loads on RVI components that may have experienced multiple aging effects.[3] LBP-08-13 68 N.R.C. at 218. Since NYS-25 was initially filed, IP2 and IP3 satisfied their initial commitment to submit an inspection plan and a RVI AMP. Entergy initially submitted the inspection plan

[3] Subsequently, New York and Riverkeeper asserted a joint contention NYS-38/RK-TC-5 that challenged the use of a commitment for RVI AMP among other issues. Since NYS-38/RK-TC-5 was admitted by the Board, IP2 and IP3 proposed a RVI AMP that establishes a program consistent with GALL and the Staff's guidance. As such, the commitment identified as deficient with respect to the RVI AMP is moot because the program is sufficiently defined by Entergy's proposal.

in September 2011, and the RVI AMP in July 2010. As a result of the Staff's review, Entergy's proposed RVI AMP was modified in some aspects to address the action items identified by the Staff. Staff Testimony at Q85, Q113-Q114, Q142-Q173. While New York has not modified its contention to address the current RVI AMP, New York's Statement of Position, supporting testimony, and exhibits raise a number of specific complaints regarding the current RVI AMP. New York's expert challenges the RVI AMP, asserting (1) the impact of synergistic aging effects are not appropriately accounted for by the program; (2) the accident analysis should be re-performed in a manner to account for any aging effects; (3) Entergy should preemptively replace components before aging effects can be detected; (4) VT-3 inspections are ineffective; and (5) the portion of the RVI AMP that makes use of fatigue program fails to conduct a sufficient error analysis. *See generally* Lahey Testimony, Ex. NYS000482.

In sum, Contention NYS-25 asserts that the RVI AMP fails to provide adequate programs for identifying and managing synergistic aging effects on the RVI components such that inspection, monitoring, and replacement will not be sufficiently robust through the period of extended operations.[4]

DISCUSSION

I. Legal and Regulatory Standards

A. Regulatory Requirements and Guidance

The Commission's requirements governing the management of aging for structures, systems and components ("SSCs") at a nuclear power plant, including reactor

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[4] As part of Contention NYS-38/RK-TC-5, New York also asserts that the RVI AMP for IP2 and IP3 is not sufficiently well defined and that it puts off resolution of important aspects of the RVI AMP until after any opportunity for a hearing through the use of license renewal commitments. *See, e.g.,* New York's SOP on NYS-38 at 4. Specifically, NYS-38/RK-TC-5 asserts that the RVI AMP is inadequate because the baffle-former bolt inspections occur during the period of extended operation and the acceptance criteria has not been set and other components may not be identified for timely replacement until the period of extended operation.

vessel internals, are set forth in 10 C.F.R. §§ 54.4 (“Scope”), 54.21(a)(3) (“Contents of Application”) and 54.29(a) (“Standards for Issuance of Renewed License”). These regulations state, in pertinent part, as follows:

> **54.4 Scope**
>
> (a) Plant systems, structures, and components within the scope of this part are--
>
> (1) Safety-related systems, structures, and components which are those relied upon to remain functional during and following design-basis events (as defined in 10 CFR 50.49 (b)(1)) to ensure the following functions--
>
> (i) The integrity of the reactor coolant pressure boundary;
>
> (ii) The capability to shut down the reactor and maintain it in a safe shutdown condition; or
>
> (iii) The capability to prevent or mitigate the consequences of accidents which could result in potential offsite exposures comparable to those referred to in § 50.34(a)(1), § 50.67(b)(2), or § 100.11 of this chapter, as applicable.
>
> (2) All nonsafety-related systems, structures, and components whose failure could prevent satisfactory accomplishment of any of the functions identified in paragraphs (a)(1)(i), (ii), or (iii) of this section.
>
> (3) All systems, structures, and components relied on in safety analyses or plant evaluations to perform a function that demonstrates compliance with the Commission's regulations for fire protection (10 CFR 50.48), environmental qualification (10 CFR 50.49), pressurized thermal shock (10 CFR 50.61), anticipated transients without scram (10 CFR 50.62), and station blackout (10 CFR 50.63).
>
> (b) The intended functions that these systems, structures, and components must be shown to fulfill in § 54.21 are those functions that are the bases for including them within the scope of license renewal as specified in paragraphs (a)(1) - (3) of this section.

54.21 Contents of application-technical information. Each application must contain the following information:

(a) An integrated plant assessment (IPA). *The IPA must*--

(1) For those systems, structures, and components within the scope of this part, as delineated in § 54.4, identify and list those structures and components subject to an aging management review. Structures and components subject to an aging management review shall encompass those structures and components--

(i) That perform an intended function, as described in § 54.4, without moving parts or without a change in configuration or properties. These structures and components include, but are not limited to, the reactor vessel, the reactor coolant system pressure boundary, steam generators, the pressurizer, piping, pump casings, valve bodies, the core shroud, component supports, pressure retaining boundaries, heat exchangers, ventilation ducts, the containment, the containment liner, electrical and mechanical penetrations, equipment hatches, seismic Category I structures, electrical cables and connections, cable trays, and electrical cabinets, excluding, but not limited to, pumps (except casing), valves (except body), motors, diesel generators, air compressors, snubbers, the control rod drive, ventilation dampers, pressure transmitters, pressure indicators, water level indicators, switchgears, cooling fans, transistors, batteries, breakers, relays, switches, power inverters, circuit boards, battery chargers, and power supplies; and

(ii) That are not subject to replacement based on a qualified life or specified time period.

* * *

(3) For each structure and component identified in paragraph (a)(1) of this section, demonstrate that the effects of aging will be adequately managed so that the intended function(s) will be maintained consistent with the CLB for the period of extended operation.

54.29 Standards for issuance of a renewed license

A renewed license may be issued by the Commission up to the full term authorized by § 54.31 if the Commission finds that:

(a) Actions have been identified and have been or will be taken with respect to the matters identified in Paragraphs (a)(1) and (a)(2) of this section, such that there is reasonable assurance

> that the activities authorized by the renewed license will continue to be conducted in accordance with the CLB, and that any changes made to the plant's CLB in order to comply with this paragraph are in accord with the Act and the Commission's regulations. These matters are:
>
> (1) managing the effects of aging during the period of extended operation on the functionality of structures and components that have been identified to require review under § 54.21(a)(1); . . . . [5]

In addition, the Commission has issued regulatory guidance pertaining to reactor vessel internals, as set forth in NUREG-1800, "Standard Review Plan for Review of License Renewal Applications for Nuclear Power Plants" ("SRP-LR") (September 2005) (Ex. NYS000195), NUREG-1801, "Generic Aging Lessons Learned (GALL) Report" (September 2005) (Ex. NYS000146A), the Staff's endorsement of MRP-227-A,[6] and Interim Staff Guidance including LR-ISG 2011-04, "Updated Aging Management Criteria for Reactor Vessel Internal Components for pressurized Water Reactors" (Ex. NYS000214).

As described in the Staff's testimony, specific guidance concerning AMPs that the Staff would find to be acceptable is provided in the GALL Report, NUREG-1801 including any updates or modifications made through Interim Staff Guidance. Staff Testimony at Q76-Q77. The GALL Report as modified and updated contains the NRC's approved set of recommendations related to "preventative actions," "mitigative actions," "condition monitoring," and "performance monitoring" as applicable to component type, material, environmental conditions, and all the applicable aging mechanisms. This is documented in a series of NRC-approved AMPs described in the GALL report, including

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[5] 10 C.F.R. § 54.29(a)(2) pertains to time-limited aging analyses, and is inapplicable to this contention concerning an aging management program.

[6] MRP-227-A, which is discussed in detail in the Staff's testimony, is a topical report analyzing pressurized water reactor vessel internals for purposes of providing the technical analysis for developing a generally applicable aging management program for pressurized water reactor.

AMP XI,M16A, "PWR Vessel Internals." Staff Testimony at Q76-Q77. In December 2010, the Staff issued GALL Report Revision 2, in which the Staff added AMP XI,M16A for RVI. (Ex. NYS000147A-D). In May 2013, the Staff issued interim Staff guidance that updated the recommended AMP in the GALL to reflect information endorsed by the Staff in MRP-227-A.

A licensee proposed AMP may be found acceptable, in one of three ways:

> 1) It may establish a program that is completely consistent with all the recommendations in the GALL Report, or
>
> 2) It may establish a program that is consistent with the GALL Report with exception(s) to certain portion(s) of the GALL Report that the applicant does not intend to implement, and/or it may state enhancements, revisions or additions to existing aging management programs that the applicant commits to implement prior to the period of extended operation to ensure that its AMP is consistent with the GALL Report AMP. Enhancements may expand, but not reduce the scope of an AMP, or
>
> 3) If an applicant's facility has specific materials, environments, aging effects and/or plant-specific operating experience for which aging cannot be effectively managed by any of the GALL Report AMPs, the applicant may develop a plant-specific program that meets the recommended format and content of an AMP as set forth in Section A.1.2.2, Aging Management Program for License Renewal, NUREG-1800, Standard Review Plan for Review of License Renewal Applications for Nuclear Power Plants.

Staff Testimony at Q60-Q68. The Staff evaluates an applicant's AMPs and its applicable exceptions and enhancements to ensure they provide reasonable assurance that the effects of aging will be adequately managed so that the in-scope system, structure or component's intended function(s) will be maintained consistent with the current licensing basis ("CLB") for the period of extended operation. *Id*.

B. Staff Witnesses

The attached testimony presents the opinion of a panel of three highly qualified witnesses: Dr. Allen Hiser, Mr. Jeffery Poehler, and Mr. Gary Stevens.

Dr. Allen Hiser, Jr., has testified previously in this proceeding on related issues and been found to be a qualified expert. Dr. Hiser is currently serves as the Senior Technical Advisor for License Renewal Aging Management in the Division of License Renewal, Office of Nuclear Reactor Regulation. He has 25 years of experience at the NRC and is serving as the chairman of the Steering Committee of the International Atomic Energy Agency (IAEA) program to develop aging management standards for international use and the International Generic Aging Lessons Learned program. In addition, he has been a member on several IAEA missions to evaluate aging management programs for international plants pursuing license renewal, and a trainer at international workshops for regulators and plants on aging management for license renewal. *See* Hiser Professional Qualifications (Ex. NRC000103). His current responsibilities include providing technical advice and assistance to the Division of License Renewal on a variety of technical, regulatory and policy issues related to aging management of nuclear power plant systems, structures, and components. His current responsibilities also include serving as a lead technical expert for aging management evaluation and assisting other NRC Staff as they implement their review of license renewal applications. Dr. Hiser holds a Bachelor of Science and Master of Science degrees in Mechanical Engineering from the University of Maryland at College Park, and a Ph.D. in Materials Science and Engineering from Johns Hopkins University. Dr. Hiser has provided testimony previously in this proceeding with respect to NYS-26, NYS-38/RKTC-38, and RK-TC-2.

Mr. Jeffery Poehler is a Professional Engineer with a Bachelor of Science and Master of Science degree in Material Science and Engineering from Johns Hopkins University. His current responsibilities include the lead technical reviewer for reactor vessel internal programs including AMP and other related issues for both reactor vessels and reactor vessel internals.

Mr. Poehler has over 22 years of experience developing and reviewing materials programs for a variety of power plant systems including nuclear and conventional power plant systems. A statement of his professional qualifications is available at Ex. NRC000.6

Mr. Gary L. Stevens has been employed by the NRC for more than five years. He is currently employed as a Senior Materials Engineer in the Vessel and Internals Integration Branch in the Division of Engineering, Office of Nuclear Reactor Regulation, U.S. Nuclear Regulatory Commission, in Rockville, Maryland. Prior to joining NRR March 2015, he was employed as a Senior Materials Engineer in the Component Integrity Branch, Division of Engineering, Office of Nuclear Regulatory Research. He received a Bachelor of Science degree in Mechanical Engineering from California Polytechnic State University in San Luis Obispo, CA, and a Master of Science degree in Mechanical Engineering from San Jose State University. He has been a participating member in the American Society of Mechanical Engineers (ASME) Code, Section XI Committees for more than 25 years. His statement of qualifications is attached hereto (Ex. NRC000227).

II. Overview of the Reactor Vessel Internals

For IP2 and IP3, the entire reactor vessel internals are generally within the scope of license renewal, except for those portions that do not meet the criteria in 10 C.F.R. § 54.21 (a)(1)(i) and (a)(1)(ii). Staff Testimony at Q9. Several components identified by New York and its witness as requiring management under the RVI AMP, however, are not covered by the RVI AMP because they because they are not passive long-lived components that require aging management or they are covered by other programs. Staff Testimony at Q9, Q12-Q13, Q256-Q257, Q259-Q261. These other programs have not been challenged by New York.

Non-passive components including control rods, and control rod drive mechanisms, are not subject to aging management under the RVI AMP and are not considered RVI components. Staff Testimony at Q9, Q12-Q13, Q259-Q261. Fuel assemblies are not subject to aging management because they are not long-lived, and nuclear instrumentation is not subject to

aging management because it is not passive. Staff Testimony at Q10. The RVI also does not include any pressure boundary components, such as the reactor pressure vessel, or the penetrations for the control rod drive mechanisms in the reactor pressure vessel closure head. *Id.*

The reactor vessel internals consist of three major assemblies: the lower core support structure, the upper core support structure, and the incore instrumentation support. Staff Testimony at Q11. These components generally provide structural support, flow distribution, or shielding. *Id.* Examples of some components that make up the three major assemblies include the core barrel (and its welds), baffle-former assembly including baffle plates, former plates, baffle-former bolts, barrel-former bolts, and baffle-edge bolts, lower core plate and support structures, clevis bolts, fuel alignment pins, thermal shield, the lower support columns, and the upper support column assemblies, the control rod guide tubes, plates, and welds the baffle-to-baffle bolts, the barrel-to-former bolts, and baffle-to-former bolts. *Id.*

The components of the RVI support the core, direct flow of reactor coolant, or provide alignment and guidance for components such as control rods or instrumentation. Staff Testimony at Q19. The major assemblies of Westinghouse-design RVI consist of the upper internals assembly that is removed during each refueling operation to obtain access to the reactor core and a lower internals assembly that can be removed following a complete core off-load. Staff Testimony at Q20. The lower internals assembly includes the core barrel, baffles and former plates, lower core plate, lower core support plate, and lower support columns. *Id.* The lower internals assembly is essentially the full height of the reactor pressure vessel, as it is supported from a ledge just below the reactor pressure vessel closure flange. *Id.* The upper internals assembly sits within approximately the upper half of the lower internals assembly, and includes the upper core plate, upper support plate, and control rod guide tube assemblies. *Id.*

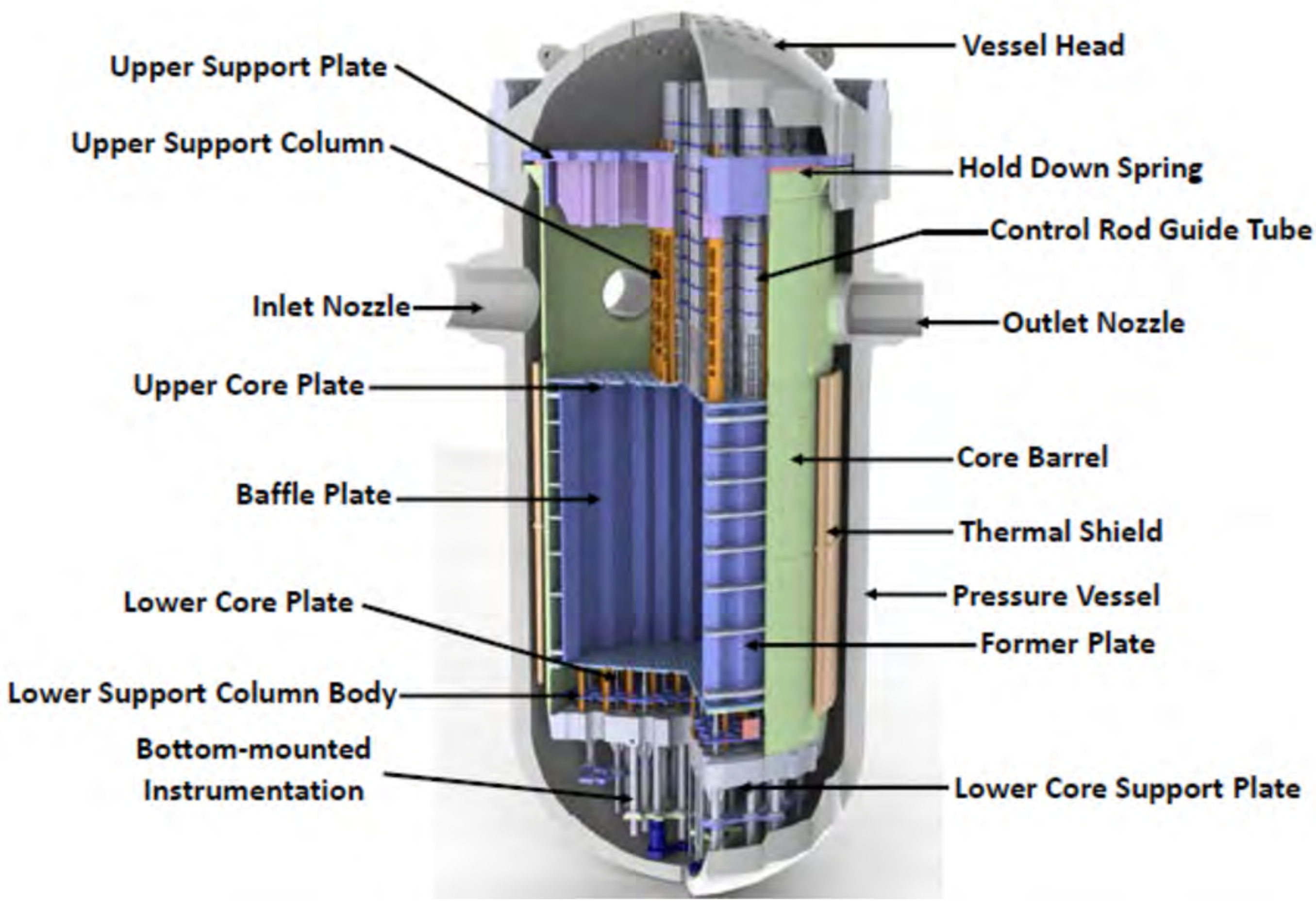

Figure 1 – Overview of Typical Westinghouse-design Reactor Vessel Internals (reproduced from update Westinghouse Figures Presented at Public Meeting on MRP-227, Rev. 1, March 31, 2015, ADAMS Accession No. ML15091A136 Ex. NRC000211)

The major sub-assemblies that constitute the upper internals assembly are the: (1) upper core plate (UCP); (2) upper support column assemblies; (3) control rod guide tube assemblies; and (4) upper support plate (USP). Staff Testimony at Q21. The primary function of the upper internals assembly is core support. Staff Testimony at Q22. The upper assembly also provides guidance to the control rods and instrumentation for temperature measurement. *Id.* During reactor operation, the upper internals assembly is preloaded against the fuel assembly springs and the internals hold down spring by the reactor vessel head pressing down on the outside edge of the USP. Staff Testimony at Q23. The USP acts as the divider between the upper plenum and the reactor vessel head and as a relatively stiff base for the rest of the

upper internals. *Id.* The upper support columns and the control rod guide tubes are attached to the USP. *Id.* The upper core plate (UCP), in turn, is attached to the upper support columns. *Id.* The UCP is perforated to permit coolant to pass from the core below into the upper plenum between the USP and the UCP. *Id.* The coolant then exits through the outlet nozzles in the core barrel. *Id.* The UCP positions and laterally supports the core by fuel alignment pins extending below the plate. *Id.* The UCP contacts and preloads the fuel assembly springs and thus maintains the fuel assemblies in contact with the lower core plate (LCP) during reactor operation. *Id.* The upper support columns vertically position the UCP and are designed to take the uplifting hydraulic flow loads and fuel spring loads on the UCP. *Id.* The control rod guide tubes are bolted to the USP and pinned at the UCP so they can be easily removed if replacement is needed. *Id.*

The control rod guide tubes, are bolted to the USP and pinned at the UCP. Staff Testimony at Q24. They, guide the control rods in and out of the fuel assemblies to control power generation. *Id.*

The major subassemblies of the lower internals assembly are the core barrel assembly, baffle-former assembly (subassembly of the core barrel assembly) and the lower support assembly. Staff Testimony at Q25.

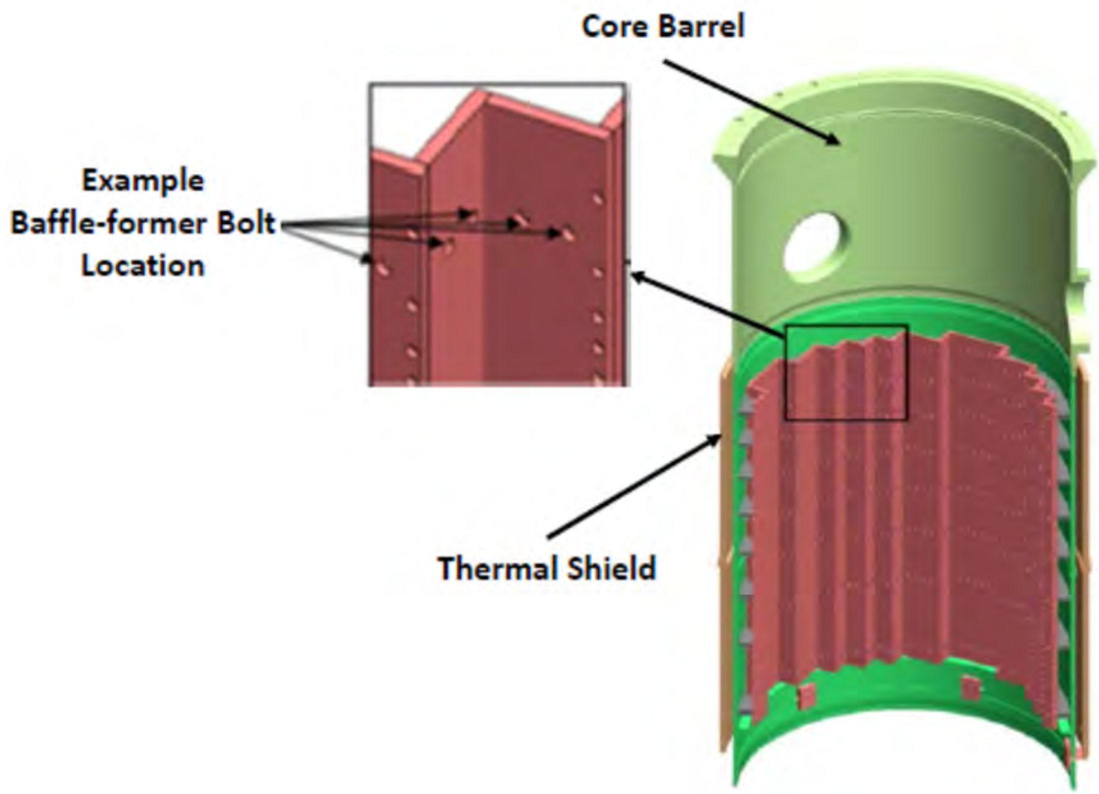

Figure 2 – Three-dimensional view of a portion of the baffle-former assembly in Westinghouse-design RVI, showing baffle-former bolt locations (reproduced from updated Westinghouse Figures Presented at Public Meeting on MRP-227, Rev. 1, March 31, 2015, ADAMS Accession No. ML15091A136, Ex. NRC000211)

The major functions of the lower internals assembly are core support and positioning, direction of flow through the core, and shielding. Staff Testimony at Q26. The fuel assemblies are supported inside the lower internals assembly on top of the lower core plate (LCP). Staff Testimony at Q27. The functions of the LCP are to position and support the core and provide a metered control of reactor coolant flow into each fuel assembly. *Id.* The LCP is elevated above the lower support casting by support columns and bolted to a ring support attached to the inside diameter of the core barrel. Staff Testimony at Q27. The support columns transmit vertical fuel assembly loads from the LCP to the much thicker lower support casting. *Id.* The function of the lower support casting is to provide support for the core. *Id.* The lower support casting is welded to and supported by the core barrel, which transmits vertical loads to the vessel through the

core barrel flange. *Id.* A large number of components are attached to the core barrel, including the baffle/former assembly, the core barrel outlet nozzles, the thermal shields, the alignment pins that engage the UCP, the lower support casting, and the LCP. *Id.* The lower radial support/clevis assemblies restrain large transverse motions of the core barrel but at the same time allow unrestricted radial and axial thermal expansion. *Id.* The baffle and former assembly consists of vertical plates called baffles and horizontal support plates called formers. Staff Testimony at Q28. The baffle plates are bolted to the formers by the baffle former bolts, and the formers are attached to the core barrel inside diameter by the barrel/former bolts. *Id.* Baffle plates are secured to each other at selected corners by edge bolts. *Id.* In addition, at IP2 and IP3, corner brackets are installed behind and bolted to the baffle plates. The baffle/former assembly forms the interface between the core and the core barrel. *Id.* The baffles provide a barrier between the core and the former region so that a high concentration of reactor coolant flow in the core region can be maintained. *Id.* A secondary benefit is to reduce the neutron flux on the vessel. *Id.* All of the RVI components located radially outboard from the core provide some shielding to the reactor pressure vessel ("RPV") by placing a metallic material between the core and the RPV. Staff Testimony at Q29. Additional neutron shielding of the reactor vessel is provided in the active core region by thermal shields attached to the outside of the core barrel. *Id.*

Dr. Lahey appears to identify the control rods and control rod drive mechanism canopy penetrations as being within the scope of the RVI AMP or indicates that they should be included. Lahey Testimony, Ex. NYS000482, at p. 12 ln. 13 – p. 13 ln. 11. While the control rods are within the scope of license renewal, they are not subject to aging management under license renewal, because they are not passive components. Staff Testimony at Q10, Q12, Q258-Q261. The control rods are not passive components because they require moving parts and changes of position to perform their intended function, therefore they do not perform their intended function "without moving parts or without a change in configuration or properties." *Id.*

These components are not within the scope of an aging management program under 10 CFR 54.21(a)(1)(i). *Id.* The control rod drive mechanism penetrations in the reactor pressure vessel closure head, are in the scope of license renewal and subject to aging management, but are addressed by the Reactor Vessel Head Penetration Inspection Program. Staff Testimony at Q13, Q263.

III. The Staff's Guidance on Reactor Vessel Internals AMP

An AMP is a program that is credited for managing the effects of aging on systems, structures and components that are within the scope of license renewal and subject to aging management review in accordance with 10 CFR 54.21(a)(1), such that the components' intended functions will be maintained as required by 10 CFR 54.21(a)(3). Staff Testimony at Q52. Although 10 C.F.R § 54.21 provides an overall framework that the license renewal application must demonstrate the adequacy of planned activities to manage the effects of aging on intended functions, it does not provide specific requirements for AMPs or specific criteria to evaluate AMPs. Staff Testimony at Q53. The Staff publishes several documents and revisions to documents that provide guidance on acceptable ways to meet the regulatory requirements, including NUREG-1801, the Generic Aging Lessons-Learned (GALL) Report, and NUREG-1800, the Standard Review Plan for Review of License Renewal Applications for Nuclear Power Plants (SRP-LR). Staff Testimony at Q54.

The GALL Report, NUREG-1801, provides, in part, aging management programs (AMPs) that the Staff has determined are acceptable for managing certain aging effects. Staff Testimony at Q55. For example, GALL Report XI.M2, "Water Chemistry," describes an effective program for water chemistry control. *Id.* The GALL Report also serves as a technical basis for the SRP-LR. *Id.* The GALL Report was initially published in 2000, and has been periodically revised based on increased knowledge on aging effects and effective aging management approaches that have come from operating experience, and the results of Staff reviews of subsequent license renewal applications. *Id.* Revision 1 of the GALL Report, NUREG-1801

was issued in 2005 (Accession Nos. ML052110005 and ML052110006) (Ex. NYS00146A-C) (together, "GALL Report Rev. 1"). *Id.* Revision 2 of the GALL Report, NUREG-1801 was issued in 2010. (Ex. NYS000147A-D) *Id.* Both Rev. 1 and Rev. 2 of the SRP-LR indicate that an applicant may reference the GALL Report in a license renewal application to demonstrate that the programs at the applicant's facility correspond to those reviewed and approved in the GALL Report, and in such a case no further Staff review is required. Staff Testimony at Q56. Although applicants are not required to use the AMPs in the GALL Report, it contains at least one acceptable way to manage aging effects for license renewal. *Id.* The GALL Report further states that an applicant may propose alternatives for Staff review in its license renewal application, but use of the GALL Report should facilitate both preparation of a license renewal application by an applicant and timely, uniform review by the NRC Staff. *Id.*

The SRP-LR, Rev. 1 and Rev. 2, provides guidance to the Staff on how to perform safety reviews of applications to renew nuclear power plant licenses in accordance with 10 C.F.R. Part 54. Staff Testimony at Q57. The principal purposes of the SRP-LR are to ensure the quality and uniformity of the Staff's review, and to present a well-defined base from which the Staff can evaluate the applicant's programs and activities for the period of extended operation. *Id.* The SRP-LR was initially published in 2000, and has been periodically revised based on increased knowledge of aging effects and effective aging management approaches from operating experience, and the results of Staff reviews of intervening license renewal applications. *Id.*

For an AMP that the applicant claims is "consistent with GALL," the license renewal application provides limited information on the AMP. Staff Testimony at Q58. The SRP-LR notes that the Staff conducts an audit and review at the applicant's facility to evaluate AMPs that the applicant claims to be consistent with the GALL Report. *Id.* NRC generic guidance for the contents of AMPs is contained in the Standard Review Plan for Review of License Renewal Applications for Nuclear Plants (SRP-LR), NUREG-1800. Appendix A.1 of the SRP-LR,

Revision 2, dated December 2010, contains guidance for plant-specific programs in Branch Technical Position RLSB-1. Appendix A.1 describes AMPs in terms of ten program elements: (1) Scope of the program, (2) Preventive Actions, (3) Parameters Monitored or Inspected, (4) Detection of Aging Effects, (5) Monitoring and Trending, (6) Acceptance Criteria, (7) Corrective Actions, (8) Confirmation process, (9) Administrative Controls, and (10) Operating Experience. Staff Testimony at Q59. NRC guidance in Chapter XI of the Generic Aging Lessons Learned (GALL) Report, NUREG-1801, provides examples of generic AMPs that have been found to be acceptable for managing aging effects for certain components, or certain aging effects for a number of components. Staff Testimony at Q60. Examples of generic AMPs include XI.M1 ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD; XI.M2 Water Chemistry; and, XI.M31 Reactor Vessel Surveillance. *Id.*

Besides using the generic AMPs found to be acceptable in the GALL Report, applicants may also propose a plant-specific AMP. Staff Testimony at Q65. An applicant proposes a plant-specific existing AMP because they have an existing program that will adequately manage the aging effects for the in-scope components. *Id.* Alternatively, applicants may propose plant specific programs where either no generic program has been previously evaluated or the limitations on the generic program exclude its use for the specific plant. *Id.* In any case, plant-specific AMPs are normally specified using the ten elements described in Appendix A.1 of the Standard Review Plan for Review of License Renewal Applications for Nuclear Plants (SRP-LR), NUREG-1800 as discussed previously. In this case, the applicant proposed a plant-specific AMP because GALL, Rev. 2, only recently identified a generic RVI AMP that would be acceptable. Staff Testimony at Q76-Q77. Although, the IP2 and IP3 proposed the RVI AMP as a plant-specific programs, the programs are consistent with GALL. Staff Testimony at Q115.

The Staff's evaluation of the adequacy of an applicant's AMP in managing certain aging effects for particular structures and components is based on its review of the ten program elements that are used to describe each aging management program. Staff Testimony at Q66.

The evaluation process is different depending on whether the license renewal application identifies the AMP as consistent with the related AMP from the GALL Report or as a plant-specific AMP. *Id.* The Staff reviews the applicant's plant-specific program for consistency with the criteria of Branch Technical Position RLSB-1 in the SRP-LR. Staff Testimony at Q59, Q68, Q235. The Staff reviews the adequacy of the program to meet the criteria of 10 CFR § 54.21(a)(3), such that the intended functions of the structures and components addressed by the AMP will be maintained. Staff Testimony at Q68.

Many AMPs simply monitor the aging effects via inspections or tests in order to identify when components will no longer be able to fulfill their intended functions. Staff Testimony at Q71. The SRP-LR refers to this type of AMP as a condition monitoring program. *Id.*

Once a component is identified as no longer being able to fulfill its intended functions during the period before the next scheduled inspection, the licensee is obligated to take corrective actions. Staff Testimony at Q71. Corrective actions could include repairing the component, replacing the component, or showing that the component will continue to fulfill its intended function despite the inspection results. *Id.* In practice, AMPs generally have acceptance criteria for inspections that would require corrective actions prior to the component approaching a condition in which it may not be able to fulfill its intended functions, so that repairs or replacement, if required, can be planned prior to loss of operability under the CLB. *Id.* Schedules for inspections and tests are designed to allow the timely detection of aging effects before a components intended functions being compromised. *Id.* This process of condition monitoring (which is followed by repair or replacement, when necessary, or additional analysis, is used to show that the components remains acceptable for service at least until the next scheduled inspection) is commonly instituted in most of the major programs for nuclear power plants and is the underlying criterion for quantifying intervals specified in most maintenance activities. *Id.*

The scope of aging management for license renewal is specified by 10 CFR § 54.4, which identifies the plant systems, structures, and components that are in the scope of license renewal. Although the regulations do not discuss long-lived and passive as criteria, these criteria are used to determine the structures and components that are subject to aging management review. Staff Testimony at Q73-Q74.

In Revision 2, the GALL Report provided an AMP, GALL Report AMP XI.M16A, "PWR Vessel Internals," and aging management review line items for the RVI. Staff Testimony at Q76-Q77. Subsequently, the Staff issued License Renewal Interim Staff Guidance LR-ISG-2011-04, "Updated Aging Management Criteria for Reactor Vessel Internal Components for Pressurized Water Reactors," on May 28, 2013 (LR-ISG). Staff Testimony at Q77-Q78. The LR-ISG replaces the guidance for the RVI AMP and the aging management review items that were provided in Revision 2 of the GALL Report, including GALL Report AMP XI.M16A, "PWR Vessel Internals," and represents the current NRC guidance for aging management of PWR RVI. *Id.* LR-ISG-2011-04 provides a revision to the AMP in GALL Report Chapter XI.M16A so that the descriptions of the ten program elements reference relevant aspects of MRP-227-A. *Id.* In addition, LR-ISG-2011-04 provided an update of the aging management review line items that are relevant for PWR RVI. *Id.*

As part of its review of MRP-227, Rev. 0, the NRC Staff identified eight action items that must be addressed by applicants or licensees (Applicant/Licensee Action Items, A/LAIs) in order to apply the methodology of MRP-227-A to aging management of the RVI at a particular plant. Staff Testimony at Q85. These plant-specific action items address topics related to the implementation of MRP-227, Rev. 0, that could not be effectively addressed on a generic basis in MRP-227, Rev. 0, and are identified in Section 4.2 of Revision 1 of the Staff's final SE approving MRP-227, Rev. 0. *Id.*

In developing the generic approved program, RVI were screened for various criteria including aging, safety significance, and redundancy among others. Staff Testimony at Q87-

Q93. After the screening and other analyses, the components were placed into one of four categories: (1) No Additional Measures, (2) Primary, (3) Expansion, and (4) Existing Programs Staff Testimony at Q93. Those components that did not screen in for any aging mechanisms, or those for which both the likelihood of failure and core damage risk was low, were assigned to the No Additional Measures category. *Id.* Other components for which one or more aging mechanisms screened in were generally categorized as either Primary or Expansion category components. *Id.* When more than one component existed with similar materials, functions, and screened-in aging mechanisms, the highest overall risk component generally became a Primary component, while lower risk components became an Expansion component. *Id.* Components determined to have high or moderate overall risk, but for which activities performed by other existing programs adequately manage the aging effect, were assigned to the Existing Programs category. *Id.* Once assigned to one of the four categories, inspection techniques were identified as appropriate based on the component and aging mechanism. *Id.*

In general, Primary components are those RVI components that have the highest susceptibility to the effects of at least one of the eight aging mechanisms. Staff Testimony at Q94. MRP-227-A generally specifies inspections of Primary components or other aging management activities, such as analyses, with most inspection required within the first two refueling outages from the start of the period of extended operation. *Id.* The Primary group also includes components that have shown a degree of tolerance to a specific aging degradation effect, but for which no component with a higher susceptibility exists or for which no component with a higher susceptibility is accessible. *Id.* Inspection findings for a Primary component that exceeds the expansion criteria will result in scheduled inspections of the Expansion components that are linked to that Primary component. *Id.*

Expansion components are those RVI components that are highly or moderately susceptible to the effects of at least one of the eight aging mechanisms, but for which functionality assessment has shown a degree of tolerance to those effects. Staff Testimony at

Q96. The implementation of aging management inspections for Expansion components will only be scheduled if the findings from the examinations of the Primary components exceed the expansion criteria or are indicated as part of the corrective actions for exceeding an acceptance criteria on a Primary Component or related operating experience from the plant or the fleet.. *Id.* Similar to the Primary Component inspection techniques, Expansion Component inspections utilize a variety of techniques, based on the applicable aging effect and the capabilities of the inspection method. Staff Testimony at Q97.

Existing Program components are those reactor vessel internals that are susceptible to the effects of at least one of the eight aging mechanisms, and existing generic or plant-specific aging management activities are adequate for managing those effects. Staff Testimony at Q98. The lower core plate is managed by the ASME Code Section XI requirements, an Existing Program. Staff Testimony at Q99.

No Additional Measures components are those RVI components for which the effects of all eight aging mechanisms are below the screening criteria. Staff Testimony at Q100. Additional components were placed in the No Additional Measures group as a result of FMECA and the functionality assessment. *Id.* No further action is required by these guidelines for managing the aging of the No Additional Measures components. *Id.*

The inspection techniques were identified based on the capability of the inspection technique to detect the expected aging effect(s) for each component. Staff Testimony at Q101. Similarly, two types of criteria were established for the inspections: acceptance criteria and expansion criteria. Staff Testimony at Q102. Acceptance criteria are used to assess the inspection findings from the Primary component inspections and determine whether corrective actions are required for the component that was inspected. Staff Testimony at Q102-Q104, Q106. Expansion criteria are used to assess the inspection findings from the Primary component inspections and then determine whether if the Expansion components require inspection. Staff Testimony at Q102, Q105-Q107.

If the acceptance criteria are exceeded, the results are entered into the plant's corrective action program for resolution, which is governed by the quality assurance requirements of Appendix B to 10 CFR Part 50. Staff Testimony at Q106. The licensee would have a number of available options or corrective actions to resolve the finding, including more detailed engineering evaluations in accordance with Section 6 of MRP-227-A, scheduling supplemental inspections, repair, replacement, or seeking a license amendment, when applicable. *Id.* Part of the corrective actions could result in inspections of additional components even if the expansion criteria were not exceeded. *Id.* If the expansion criteria are exceeded, then MRP-227-A specifies that the inspection of the Expansion component associated with the Primary component must be completed within a specified time frame. Staff Testimony at Q107. For example, if a two-inch crack is found during the EVT-1 inspection of the lower core barrel girth weld, the lower core barrel cylinder axial welds must be inspected by the end of the next refueling outage. *Id.*

Entergy's RVI AMP manages aging by condition monitoring through regularly scheduled inspections of components that have been identified as leading indicators for potential aging effects. Staff Testimony at Q112. Components are inspected using a variety of inspection techniques including visual examination, ultrasonic examination, and physical measurements. Staff Testimony at Q80. The RVI AMP is a sampling-based program; in other words, every RVI component is not inspected. *Id.* Rather, components are selected for inspection based on a combination of susceptibility to aging effects and the risk of core damage if aging effects were to occur in a particular component. *Id.*

IV. Entergy's RVI AMP Adequately Manages the Aging Effects During the Period of Extended Operation

New York's asserts that the IP2 and IP3 RVI AMP will not be able to manage aging effects because Entergy "relies on detection of cracks or other aging effects prior to part repair or replacement." New York SOP at 31. New York's expert emphasizes that the components

are experiencing multiple aging effects that will interact synergistically such that aging will exceed the sum of each effect individually. Lahey Testimony at p. 15 ln. 16 – p. 16 ln. 10. New York also disputes whether the VT-3 inspection techniques are sufficient to identify aging effects during the scheduled inspections. New York SOP at 28; Lahey Testimony at p. 51 lns. 1-15. New York and its expert unreasonably and without support dismiss the condition monitoring program and would essentially require a schedule for the regular replacement for all the RVI components regardless of detectable aging effects. *See* Lahey Testimony at p. 54 lns. 3-8; p. 57 lns. 2-9. As Dr. Hiser and Mr. Poehler explain, this kind prophylactic replacement of components is not necessary for an effective aging management program and fails to consider the significance of components and their redundancy. Staff Testimony at Q180, Q235-Q243.

A. <u>The RVI Is an Appropriately Designed Condition Monitoring Program</u>

The IP2 and IP3 RVI AMP consists of a program description, describing the ten elements of the AMP, and an inspection plan. Staff Testimony at Q114. The program description was initially submitted on July 14, 2010 and was revised in the letter dated February 17, 2012. *Id.* The inspection plan contains tables specifying the inspections for the Primary, Expansion, and existing Programs components, and tables contain the expansion acceptance criteria for these components. *Id.* The Inspection Plan also contains Entergy's proposed resolution of the applicant/licensee action items (A/LAIs) from the Staff's final safety evaluation of MRP-227, Rev.0. *Id.* The inspection plan was initially submitted on September 28, 2011, and a revised version consistent with MRP-227-A was submitted on February 17, 2012. *Id.* Some modifications to Entergy's proposed responses to the action items were made in various responses to requests for additional information. *Id.*

Entergy's RVI AMP is consistent with the Staff guidance because it is contains the ten program elements of Chapter XI.M16A, "PWR Internals," as defined in LR-ISG-2011-04. Staff Testimony at Q115. These elements are (1) scope, (2) preventive actions, (3) detection of aging effects, (4) parameters monitored and inspected, (5) monitoring and trending, (6)

acceptance criteria, (7) corrective actions, (8) confirmation process, (9) administrative actions, and (10) operating experience. Staff Testimony at Q119. The AMP is based on MRP-227-A and implements all the Primary, Expansion, and Existing Programs inspections specified for Westinghouse-design RVI in MRP-227-A. Staff Testimony at Q121. The components to be inspected, inspection methods, and inspection schedules are consistent with those recommended in MRP-227-A. Staff Testimony at Q120.

One important concept missing from Dr. Lahey's analysis is lack of credit for Primary/Secondary Water Chemistry Program. Staff Testimony at Q122. LR-ISG-2011-04 states that an RVI AMP may rely on PWR water chemistry control to prevent or mitigate aging effects that can be induced by corrosive aging mechanisms (e.g., loss of material induced by general corrosion, pitting corrosion, crevice corrosion, or stress corrosion cracking or any of its forms (SCC, PWSCC, or irradiation-assisted SCC)). Staff Testimony at Q122-Q123. Entergy's RVI AMP credits the Water Chemistry Control Program – Primary and Secondary for minimizing the potential for loss of material due to general corrosion, pitting corrosion, and crevice corrosion, and for minimizing the potential for SCC, irradiation-assisted SCC, and PWSCC. Staff Testimony at Q123. The Water Chemistry Program is important because it minimizes the concentrations of impurities such as chloride and fluoride ions and dissolved oxygen, thus minimizing the potential for aging mechanisms such as SCC. *Id.* Water chemistry is largely responsible for the very low incidence of SCC in pressurized water reactors RVI over the first forty years of operation. *Id.* Entergy also implemented a low-leakage core design for IP2 and IP3 prior to thirty calendar years of operation, which reduced the potential for irradiation-driven aging mechanisms such as irradiation-assisted SCC, irradiation embrittlement, void swelling, and ISR. Staff Testimony at Q124, Q147.

Entergy's RVI AMP is implementing the monitoring and inspecting criteria in accordance with the Staff's guidance. Staff Testimony at Q130. Of note, the applicant's AMP meets the required inspection coverage from the tables in Section 4.0 of MRP-227-A, which contains

criteria for the overall percentage of the total component or redundant population that must be inspected (75%) to claim credit for completing the inspection. *Id.* Therefore, the Staff found that this recommendation of LR-ISG-2011-04 is met. *Id.* Based on its review of the LRA, the Staff found the applicant's description of the "monitoring and trending" program element was acceptable because the AMP is consistent with the criteria of LR-ISG-2011-04 Section XI.M16A for this element. *Id.*

The Staff found Entergy's AMP was consistent with the Staff's guidance for corrective actions because detected conditions not satisfying the examination acceptance criteria will be processed through the plant's corrective actions programs. *Id.* Additionally, Entergy's AMP includes provisions to use the methodologies in Section 6 of MRP-227-A or other acceptable alternatives to analytically disposition unacceptable conditions, which is consistent with LR-ISG-2011-04. Staff Testimony at Q130, Q134. MRP-227-A, Section 6 describes methods for evaluating conditions such as cracks detected during inspections of RVI. These methods cover supplemental non-destructive evaluation, crack growth rates to be assumed, and techniques for evaluating the structural stability of cracks.[7] *Id.*

The RVI Inspection Plan is a document describing how the inspections and other aging management activities of Entergy's RVI AMP will be implemented. Staff Testimony at Q141. The RVI AMP description only describes the ten elements of the RVI AMP to demonstrate compliance with the GALL Report, while the RVI Inspection Plan contains the details of the inspections of the Primary, Expansion, and Existing Programs components for IP2 and IP3. The RVI Inspection Plan also describes how Entergy proposed to resolve each Applicant/License Action Item ("A/LAI"). Staff Testimony at Q142. Only A/LAI 1,2,3,5, 7 and 8

[7] WCAP-17096-NP is a topical report intended by EPRI to be used in conjunction with MRP-227-A, which provides more detailed methods for performing engineering evaluation of conditions detected during MRP-227-A RVI inspections which do not meet the examination acceptance criteria. Staff Testimony at Q134. In its June 14, 2012 response, the applicant indicated it intended to use the guidance of WCAP-17096-NP. *Id.*

are applicable to IP2 and IP3.[8] Staff Testimony at Q143. A/LAI 1 requires an applicants or licensees to assess its design and operating history and demonstrate the applicability of MRP-227-A to its plant. Staff Testimony at Q145-Q148. Entergy used guidance issued by the EPRI MRP in letter MRP 2013-025 to resolve this issue. *Id.* MRP 2013-025 provides quantitative criteria based on core power density (related to fuel management) and geometry. *Id.* These criteria were approved by the NRC Staff in a Safety Assessment – U.S. Nuclear Regulatory Commission, Office of Nuclear Reactor Regulation, "Evaluation of WCAP-17780-P, "Reactor Internals Aging Management MRP-227-A Applicability for Combustion Engineering and Westinghouse Pressurized Water Reactor Designs," and MRP-227-A, Applicability Guidelines for CE and Westinghouse Pressurized Water Reactor Designs," November 7, 2014. *Id.* If the plant meets the quantitative criteria, it can be reasonably assured that the assumptions made in developing MRP-227-A regarding neutron fluence and internal metal temperature of RVI components adequately represent the plant. *Id.* MRP 2013-025 also provides guidance for assessing whether a plant has components with 20% or greater cold work. *Id.* Entergy demonstrated that IP2 and IP3 met the core power density and geometry related criteria for Westinghouse-design RVI, and also demonstrated that IP2 and IP3 have no components with 20% or greater cold work. *Id.* In addition to the above criteria, Entergy also demonstrated the three basic applicability criteria from MRP-227-A, Section 2.4, are met which are: (1) 30 years of operation with high leakage core loading patterns (fresh fuel assemblies loaded in peripheral locations) followed by implementation of a low-leakage fuel management strategy for the remaining 30 years of operation; (2) base load operation, i.e., typically operates at fixed power levels and does not usually vary power on a calendar or load demand schedule; and (3) no design changes beyond those identified in general industry guidance or recommended by the original vendors. *Id.* Since Entergy adequately addressed the two factors for which the Staff

[8] A/LAI 4 and A/LAI 6 only apply to Babcock & Wilcox design RVI, and therefore, are not applicable to IP2 and IP3. Staff Testimony at Q144.

determined that additional plant-specific information was necessary to verify applicability of MRP-227-A to IP2 and IP3 – (1) cold work induced stress; and, (2) fuel management – by confirming that IP2 and IP3 comply with the criteria defined in the guidance document, the Staff determined Entergy adequately resolved A/LAI 1. *Id.*

A/LAI 2 requires an applicant or licensee to review all the RVI components within the scope of license renewal at its plant, and determine whether all these components correspond to a generic component for the applicable vendor design. Staff Testimony at Q149-Q152. If any components are found within the scope of license renewal for the plant that do not have a corresponding generic component, the plant-specific RVI program must be modified to include these plant-specific components. *Id.* Entergy reviewed the IP2 and IP3 components within the scope of license renewal against the generic list of components for Westinghouse-design RVI in technical report MRP-191 Revision 0, "Materials Reliability Program: Screening, Categorization and Ranking of Reactor Internals of Westinghouse and Combustion Engineering PWR Designs." *Id.* Entergy determined there were no IP2 and IP3 components that were not equivalent to a generic component for Westinghouse-design RVI, although there were some components with differences in the specified material. *Id.* The material differences were generally the use of a different grade of austenitic stainless steel than assumed for the generic component. *Id.* Entergy evaluated the components made from different materials to determine if any changes in aging management were necessary, and determined no changes in aging management were necessary. *Id.* The Staff found the applicant's response to A/LAI 2 acceptable because the applicant verified that all of the IP2 and IP3 components within the scope of LR are covered by MRP-191, which the Staff confirmed, and the Staff determined via an audit of the applicant's evaluation of the material differences that the applicant's evaluation was acceptable. *Id.*

A/LAI 3 requires an applicant or licensee to perform a plant-specific analysis to justify the adequacy of its plant-specific programs. Staff Testimony at Q153-Q156. For Westinghouse-

design RVI such as those of IP2 and IP3, the only component to which A/LAI 3 applies is the guide tube support pins (split pins). *Id.* Entergy's plant-specific program addressed SCC of split pins by replacement with new pins made from SCC-resistant materials. *Id.* For IP2, the split pins were replaced in the 1990's with Alloy X-750 pins. *Id.* This is the same alloy as the original split pins, but the replacement pins had a different heat treatment that makes them more resistant to SCC. *Id.* The IP3 split pins were replaced in 2009 with cold-worked Type 316 stainless steel pins, which operating experience has shown are highly resistant to SCC. *Id.* Entergy also indicated that it intended to replace the IP2 pins with cold-worked Type 316 stainless steel pins in 2016. *Id.* To address Staff concerns regarding the possibility of SCC of the IP2 pins during the time prior to replacement, Entergy performed statistical modeling based on operating experience that showed a very low probability of split pin failures prior to the planned replacement date. *Id.* Entergy also made a commitment to replace the IP2 pins during the 2016 refueling outage. *Id.* The Staff found that the applicant's plant-specific program for split pins was acceptable because the existing pin replacement date has been justified. *Id.* The date is formalized through a license commitment that will be made a requirement through license condition on issuance of the license renewal. *Id.* The split pins resistance to SCC will be enhanced through the use of improved materials. *Id.*

A/LAI 5 requires an applicant or licensee to identify plant-specific acceptance criteria to be applied when performing the physical measurements required by MRP-227-A. Staff Testimony at Q157-Q160. The only component for which physical measurements are specified for Westinghouse-design RVI such as those of IP2 and IP3 is the hold-down spring. *Id.* Entergy stated for IP2 and IP3 that the acceptance criteria are a function of spring height as a function of time relative to the required hold-down force. *Id.* The decrease in the hold-down spring height is assumed to occur linearly over time. *Id.* The approach linearly interpolates the required minimum spring height at the time of measurement between the spring height at startup and the minimum spring height at 60 years. *Id.* If the first spring height measurement is

less than the required height, successive (additional) measurements of spring height will be performed or the hold-down spring will be replaced. *Id.* Based on the its review, the Staff determined that the assumption of a linear decrease in hold-down spring height as a function of time is conservative for this analysis. *Id.* [redacted] *Id.* The Staff found Entergy's response to A/LAI 5 acceptable because Entergy provided a plant-specific acceptance criterion for the remaining compressible height or hold-down force of the hold-down spring. *Id.*

A/LAI 7 requires an applicant or licensee to perform a plant-specific analysis of cast austenitic stainless steel ("CASS") RVI components to demonstrate the components will remain capable of performing their intended functions through the end of plant life. Staff Testimony at Q161-Q164. The analyses must account for the potential loss of fracture toughness of the components due to both thermal embrittlement and irradiation embrittlement, as applicable. *Id.* The requirement to perform the functionality analysis is focused on components that require aging management according to the guidance of MRP-227-A. *Id.*

Entergy identified the only components requiring a response to A/LAI 7 for IP2 and IP3 are the lower support columns. *Id.* Only the upper portion of the lower support columns, known at the column cap, is made from CASS. *Id.* The lower support columns are an Expansion

component in MRP-227-A, and the associated (linked) Primary component is the control rod guide tube lower flange welds. *Id.* Figure 3 shows a detail of the construction of typical lower support columns. *Id.*

Figure 4-34
Typical Westinghouse core support column. Core support column bolts fasten the top of the support column to the lower core plate

Figure 3 – Lower Support Columns, from MRP-227-A Figure 4-34 (Ex. NRC000114B)
The Staff's final SE of MRP-227, Rev. 0 described three possible methods for applicants or licensees to perform functionality analyses of CASS components: (MRP-227, Rev. 0 SE at p. 23 Ex. ENT000230)

Entergy screened the lower support columns for IP2 and IP3 for thermal embrittlement susceptibility using the delta ferrite content calculated from the actual chemical composition of the columns, which was obtained from plant construction records. *Id.* The lower support columns for IP2 and IP3 are made from Type CF-8 stainless steel, which is a low-molybdenum content grade of cast stainless steel. *Id.* Low-molybdenum cast austenitic stainless steel ("CASS") grades are less susceptible to thermal embrittlement than high-molybdenum CASS grades such as Type CF-8M. *Id.* Entergy determined the lower support columns at IP2 and IP3 were not susceptible to thermal embrittlement. *Id.* Entergy used the criteria from "License Renewal Issue No. 98-0030, *Thermal Aging Embrittlement of Cast Stainless Steel*

Components." *Id.* These screening criteria would allow low-molydenum CASS with delta ferrite content of 20 % or less to be screened out (not susceptible) to thermal embrittlement. *Id.* Since portions of the lower support column will have neutron fluence at the end of life greater than $1x10^{17}$ n/cm^2, the Staff did not accept Entergy's screening. *Id.* Entergy did determine the columns are susceptible to irradiation assisted embrittlement, using a screening criterion of $6.7x10^{20}$ n/cm^2 (1 dpa) as the fluence above which the cast austenitic stainless steel ("CASS") materials is susceptible to irradiation embrittlement. *Id.* However, using the delta ferrite values for the IP2 and IP3 lower support columns, the Staff applied alternate interim screening criteria developed by the NRC Staff (NRC, "NRC Position on Aging Management of CASS Reactor Vessel Internal Components," June 11, 2014, ADAMS Accession No. ML14163A112. Ex. NRC000201), and determined that the IP2 and IP3 lower support columns could be screened out for thermal embrittlement. *Id.* According to the Staff's updated criteria, low-molybdenum statically cast CASS with a ferrite content less than 15 percent can be screened out for thermal embrittlement and any synergistic effects of thermal embrittlement and irradiation embrittlement, and is only susceptible to irradiation embrittlement at fluences greater than 1×10^{21} n/cm^2 (1.5 dpa). *Id.* This ferrite value represents a reduction in the ferrite content from the screening values from the Staff's May 19, 2000, letter to accommodate possible combined effects of thermal and neutron embrittlement. *Id.* The ferrite content of the IP2 and IP3 column caps meets this criterion because all the materials used to fabricate the column caps have calculated ferrite content less than 15 percent. *Id.*

In addition to screening for TE and IE, Entergy also provided information for the lower support columns from original fabrication demonstrating that there is a low probability that the columns have original fabrication defects. *Id.* Specifically, a liquid penetrant examination and radiography were performed during original fabrication, and no surface breaking defects were found. *Id.* Entergy also determined that service-induced cracking was very unlikely because (1) The operating stresses in the columns are too low to cause irradiation-assisted SCC at the

expected neutron fluence levels for the columns, and (2) Entergy performed fatigue analyses of the columns showing cumulative usage factors adjusted for the effects of the reactor water environment (CUF_{en}) of less than 1.0 for the life of the plant, which means that fatigue crack initiation is not expected during the life of the plant. *Id.* In response to NRC Staff concerns regarding whether the associated primary component for the lower support columns, the control rod guide tube lower flange welds, was a good predictor for irradiation assisted SCC and irradiation embrittlement, Entergy added an additional plant-specific Primary component linked to the lower support columns. *Id.* This component is the core barrel lower girth welds, which are exposed to similar neutron fluence levels as the upper portion of the lower support columns and also are susceptible to irradiation-assisted SCC and irradiation embrittlement. *Id.* Although the control rod guide tube lower flange welds screened in for irradiation embrittlement and irradiation-assisted SCC, they experience much lower neutron fluence levels so are not as good of a predictor for these aging mechanisms for the lower support columns. *Id.* The Staff found that Entergy adequately addressed A/LAI 7 based on the following: (1) Entergy evaluated the risk-significant cast austenitic stainless steel ("CASS") components of the RVI (the lower support columns (column caps)); (2) Entergy screened the column caps for thermal embrittlement and irradiation embrittlement using plant-specific materials data and determined that the column caps are not susceptible to thermal embrittlement (and the Staff confirmed the results of the screening using its own screening criteria); (3) Entergy provided information on fabrication non-destructive examination ("NDE") demonstrating that pre-existing flaws are unlikely to exist in the column caps; (4) Entergy provided information on the expected stresses and neutron fluence for the column caps that demonstrated that service-induced cracking due to irradiation-assisted SCC is unlikely; and (5) Entergy modified its RVI Inspection Program to include a link to a lead component that is an appropriate predictor of irradiation-assisted SCC and irradiation embrittlement for the column caps, with an appropriate schedule for performing the Expansion inspection if necessary. *Id.* The Staff found that the information provided by

Entergy provides reasonable assurance that the functionality of the column caps will be maintained during the period of extended operation for IP2 and IP3. *Id.*

A/LAI 8 requires applicants or licensees to make a submittal for NRC review and approval to credit their implementation of MRP-227-A as an AMP for the RVI components at their facility. Staff Testimony at Q165-Q168. Although Entergy submitted the LRA for IP2 and IP3 before the issuance of the final SE for MRP-227, Rev. 0, Entergy chose to provide the information for all five items. The items are summarized as follows:

1. An AMP for the facility that addresses the 10 program elements as defined in NUREG-1801, Revision 2, AMP XI.M16A.

2. An RVI Inspection Plan addressing all the applicable A/LAI's.

3. A Final Safety Analysis Report (FSAR) supplement containing a summary description of the RVI AMP and any TLAAs related to RVI. This is required for all AMPs and TLAAs by 10 CFR 54.21(d).

4. Any changes to technical specifications required to manage aging of the RVI, as required by 10 CFR 54.22.

5. Identify all analyses in the current licensing basis that are TLAAs related to RVI, as required in general for TLAAs by 10 CFR 54.21(c)(1). A/LAI 8 also stated that for those cumulative usage factor (CUF) analyses that are TLAAs, the applicant may use the PWR Vessel Internals Program as the basis for accepting these CUF analyses in accordance with 10 CFR 54.21(c)(1)(iii) only if the RVI components in the CUF analyses are periodically inspected for fatigue-induced cracking in the components during the period of extended operation. The periodicity of the inspections of these components shall be justified to be adequate to resolve the TLAA. Otherwise, acceptance of these TLAAs shall be done in accordance with either 10 CFR 54.21(c)(1)(i) or (ii), or in accordance with 10 CFR 54.21(c)(1)(iii) using the applicant's program that corresponds to the GALL Report, Revision 2, AMP X.M1, "Metal Fatigue of Reactor Coolant Pressure Boundary Program." To satisfy the evaluation requirements of ASME Code Section III, Subsections NG-2160 and NG-3121, the existing fatigue CUF analyses should include the effects of the reactor coolant system water environment.

Id. Entergy provided an RVI AMP, RVI Inspection Plan, and FSAR Supplement describing the RVI AMP, thus meeting the requirements of Items 1-3 of A/LAI 8. *Id.* Entergy stated that no TS changes were required related to the RVI AMP, thus Item 4 was not applicable. *Id.* For Item 5, the LRA contained a TLAA for fatigue of RVI, with CUFs reported in the LRA. *Id.* The original

RVI fatigue CUFs did not account for the effects of the reactor water environment. *Id.* Entergy resolved Item 5 via a license renewal commitment to update these CUFs to account for the effects of the reactor water environment, and management of the fatigue TLAAs for RVI using its Fatigue Monitoring Program, which is consistent with the guidance of the GALL Report, Revision 2, AMP X.M1, "Metal Fatigue of Reactor Coolant Pressure Boundary Program." *Id.* Entergy's calculation of the environmentally adjusted CUFs, or CUF_{en} values, will use methods consistent with those that are acceptable according to the GALL Report, Revision 2, Section X.M.1, "Fatigue Monitoring." *Id.* The Staff found Entergy acceptably resolved A/LAI 8 because it provided all the required information items. *Id.* For Item 5, the Staff found Entergy's resolution acceptable based on its commitment to recalculate the RVI CUFs using NRC-approved methodology to account for environmental effects, and also because it will use its Fatigue Monitoring Program, which is consistent with the Staff's guidance, to manage the fatigue TLAA as specified for Item 5, when fatigue is not managed via inspections. *Id.*

For the Entergy RVI AMP, Table 5-2 of the RVI Inspection Plan specifies the inspection methods for the Primary components. Staff Testimony at Q169. The Staff found these inspections were all consistent with the Primary component inspections specified in MRP-227-A for Westinghouse-design RVI. *Id.* Table 5-3 of the RVI Inspection Plan specifies the inspection methods for the Expansion components. Staff Testimony at Q170. The inspections of the Existing Programs components under Entergy's AMP are listed in Table 5-4 of the RVI Inspection Plan. Staff Testimony at Q171. The Staff found these inspections are consistent with the Expansion and Existing Programs inspections specified in MRP-227-A for Westinghouse-design RVI. *Id.* The Primary, Expansion and Existing Programs components, inspection scope, methods, and schedules that are specified under Entergy's RVI AMP are identical to those specified in MRP-227-A for generic Westinghouse RVI. Staff Testimony at Q172. The Staff determined that the RVI AMP submitted by Entergy for IP2 and IP3 is

consistent with the Staff's guidance and will adequately manage the aging effects during the period of extended operation. *Id.*

B. The Synergistic Effects of Aging Are Effectively Monitored Such That Timely Analysis, Repair, or Replacement Will Occur

Dr. Lahey asserts that the impact of synergistic aging effects is not appropriately accounted for by the RVI AMP. Lahey Testimony at p. 15 ln. 16 – p. 16 ln. 10; New York SOP at 28. Dr. Lahey asserts that some forms of aging are not readily detectable and the effects will be greater than the sum of each effect when combined. Lahey Testimony at p. 60 ln. 19 –p. 61 ln. 21; New York SOP at 28. This concern is misplaced because the RVI AMP is a condition monitoring program. As explained above, inspections are scheduled and any cracks, defects, or flaws that are identified are compared against both the acceptance criteria for the component and the expansion criteria that would trigger additional mandatory inspections of Expansion Components. Staff Testimony at Q183-Q205. Inspections, unlike analysis, fully consider any impacts from synergistic aging effects because they measure the impact and can trend any effect or combination of effects, when necessary. *Id.*

As explained above, the acceptance and expansion criteria are set at appropriately conservative levels to ensure actions are taken prior to the loss of any intended functions. Staff Testimony at Q179. Dr. Hiser and Mr. Poehler explained that the RVI AMP has divided components into various categories for inspection including Primary Components that will be inspected, Expansion Components that will be inspected depending on the results from the Primary Component inspection, Existing Programs that are currently being inspected or managed under an existing inspection program, and No Further Action that includes components not subject to an aging effect or have low safety significance and functional redundancy. Staff Testimony at Q93-Q107. While Dr. Lahey argues that synergistic effects are likely to be worse than the effects individually, the Staff's experts explain that the most recent research tends to indicate that synergistic aging effects can off-set each other. Staff Testimony

at Q200. For example, Dr. Hiser and Mr. Poehler highlight the offsetting effects of irradiation assisted creep and embrittlement. Staff Testimony at Q50. More importantly, the inspections identify aging in the components regardless of the aging effect or if more than one aging effect exists. Staff Testimony at Q183-Q205. Thus, the RVI AMP is sufficiently robust to adequately manage the possibility of synergisms between aging effects that could result in more rapid aging of certain components.

C. The Appropriate Inspection Techniques Were Selected for the Scheduled Inspections

Dr. Lahey appears to assert that only VT-3 inspections are being used to inspect RVI components and that VT-3 is an inadequate inspections technique. Lahey Testimony at p. 62 lns. 1-15; New York SOP at 28. New York's expert is mistaken on both issues.

First, the RVI AMP utilizes multiple inspection techniques including ultrasonic techniques ("UT"), enhanced visual inspections ("EVT-1"), and visual techniques – VT-1 and VT-3. Staff Testimony at Q80. These inspection techniques are selected based on the component, its materials, the aging effects, and the safety significance of the component including any built-in redundancy. Staff Testimony at Q80, Q95, Q101. For example, UT inspections are scheduled for the baffle former bolts. Staff Testimony at Q80. Although Dr. Lahey is concerned that VT-3 would not be able to detect the aging effects and specifically identifies the baffle-former bolts as a component unsuited for visual inspections, these bolts are actually inspected by UT and New York would has not raised any challenge to the adequacy of the UT inspection techniques. New York and its expert did not challenge the use of EVT-1 or VT-1 techniques. For example, EVT-1 and VT-1 are used on several components including lower core barrel cylinder girth welds and upper core barrel cylinder girth welds. *Id.* Thus, the only issue remaining is the adequacy of VT-3 inspections, Dr. Lahey rejects VT-3 inspections out-of-hand because he believes that those inspections do not necessarily identify every possible instance of potential aging. Lahey Testimony at p. 62 lns. 1-15. The examples cited by Dr. Lahey consist primarily of bolts and

split pins. Lahey Testimony at P. 55 ln1 – p. 58 ln. 14. However, baffle-former bolts are being inspected by UT and the split pins have been scheduled for replacement even before New York intervened. Staff Testimony at Q155, Q181. The clevis insert bolts are tightly retained due to their design. Staff Testimony at Q282-Q295. They have a welded in place retaining bar that precludes their relocation, aids in their ability to maintain their function even during failure, and the built-in redundancy of function from multiple clevis bolts. *Id.*

While Dr. Lahey's concerns regarding VT-3 inspections seems to be a preference for alternative inspections without fully analyzing whether the VT-3 inspection is enough. VT-3 inspections are fully evaluated and endorsed by the ASME code. Staff Testimony at Q108-Q109. In the case of RVI components, the VT-3 inspection technique was selected for the appropriate components and uses sufficiently conservative acceptance criteria. Staff Testimony at Q244-Q252. Thus, VT-3 inspection technique along with the other regularly scheduled inspections using other techniques and the Primary and Secondary Water Chemistry Program adequately manage the aging effects such that the intended functions will be maintained during the period of extended operation. Thus, the RVI AMP is adequate to manage the aging effects during the period of extended operation.

V. <u>A New Accident Analysis Is Outside the Scope of License Renewal</u>

One of the primary errors infecting each of New York's assertions regarding NYS-25 is Dr. Lahey's argument that the Entergy should perform new accident analysis to ensure that the RVI components are capable of withstanding shock loads in their embrittled and fatigued state. Lahey Testimony at p. 68 ln 4 – p. 69 ln. 4; New York SOP at 40-41. New York assertions expand the scope of this argument to include requirements to conduct accident analysis based on hazards that are not part of the current license basis ("CLB") or design-basis accidents. These assertions directly challenge the Commission's regulations, are outside the limited scope of the license renewal proceeding, and are not material to the findings the NRC must make.

The Commission's regulations governing the scope of license renewal are clear. The Commission's regulations in 10 C.F.R. Part 54 limit the scope of a license renewal proceeding to the specific matters that must be considered for the license renewal application to be granted. These standards, along with other regulations in 10 C.F.R. Part 54, establish the scope of safety issues that may be considered in a license renewal proceeding.

The Commission has provided guidance for license renewal adjudications regarding which safety issues properly fall within or beyond its license renewal requirements. Importantly, the Commission has found it unnecessary to include a review of issues already monitored and reviewed as part of ongoing regulatory oversight processes. *Florida Power & Light Co.* (Turkey Point Nuclear Generating Plant, Units 3 & 4), CLI 01 17, 54 NRC 3, 8-10 (2001).

The Commission clearly indicated that its license renewal safety review focuses on "plant systems, structures, and components for which current [regulatory] activities and requirements may not be sufficient to manage the effects of aging in the period of extended operation." *Turkey Point*, CLI 01 17, 54 NRC at 10 (quoting 60 Fed. Reg. at 22,469). For example, the Commission has held that emergency planning issues are not within the scope of license renewal because "[e]mergency planning is, by its very nature, neither germane to age-related degradation nor unique to the period" of extended operation. *Millstone*, CLI-05-24, 62 NRC at 560-61. Similarly, a new accident analysis is also outside the scope of the license renewal proceeding and not material to the findings that the Staff must make.

With respect to the safety review, the Commission has provided significant guidance on the structures, systems, and components within the scope of license renewal, as well as the intended functions of those structures, systems, and components that require aging management review in CLI-10-14. Therein, the Commission stated, "aging management review for license renewal does not focus on all aging-related issues," but rather, on "structures and components that perform 'passive' intended functions" that are of "principle importance to safety." The Commission explained that 10 C.F.R § 54.4(a)(1)-(3) defines the general scope of

license renewal safety review, and further stated that 10 C.F.R. § 54.29, which lists the standards for issuance of a renewed license, does not expand the scope of license renewal aging management review beyond the intended functions outlined in 10 C.F.R. § 54.4.

New York's claims that Entergy should conduct new or re-perform old accident analysis utilizing different or alternative input parameters are not material to the contention because CLB compliance is not within the scope of a license renewal proceeding pursuant to 10 C.F.R. §§ 54.30(b). The Commission has made clear that if a petitioner has safety concerns that go to the adequacy of the plant's current licensing basis, the proper forum to raise such a challenge is a petition under 10 C.F.R. § 2.206 for NRC action on the license. Thus, to the extent New York believes that current accident analysis that forms part of the IP2 and IP3 CLB is inadequate, its remedy is to petition the NRC to act under 10 C.F.R. § 2.206 rather than try to raise it as part of the license renewal proceeding.

Moreover, the scope of a license renewal proceeding is narrowly tailored. The issue to be decided during a license renewal proceeding is whether the aging effects will be adequately managed such that the intended function will be maintained through the period of extended operation. 10 C.F.R. § 54.29. The regulations do not contemplate conducting new accident analyses or reanalyzing design basis accidents with new inputs. The Commission through license renewal rulemaking established a different method for determining whether a renewed license should be issued. 10 C.F.R. § 50.49. As Dr. Hiser and Mr. Poehler explain, the Staff uses the AMP to determine that RVI components (in this case) are able to perform their intended functions through inspections designed to identify the components prior to any aging effect including embrittlement, fatigue, or combinations of aging mechanisms compromise the components intended functions. Staff Testimony at Q206-Q234. Each component has acceptance criteria based on maintaining the component's function during normal operations and design basis accidents. *Id.* As such the condition monitoring program established by the RVI AMP, the water chemistry program, and the fatigue monitoring program serve to reduce

aging effects and assure that components are identified in a timely manner such that additional analysis, repair, or replacement can occur as necessary. *Id.*

Finally, applicants are not required to perform their design basis accident analysis again as part of license renewal. *Id.* New York argues that IP2 and IP3 should consider the new seismic hazards that were submitted in response to the 10 C.F.R. § 50.54(f) request after the Fukushima Dai-ichi accident. New York SOP at 40-41, 45-46. This new seismic hazard is not part of the licensing basis for IP2 or IP3. Entergy continues to develop seismic PRAs in response to the Staff's request and the Staff continues to review the submissions. The new seismic hazard is not part of the IP2 or the IP3 licensing basis, and thus, not material to Staff's findings. *Turkey Point*, CLI 01 17, 54 NRC at 10 (quoting 60 Fed. Reg. at 22,469). As the Commission has explained, the purpose of the license renewal review is not to revisit every aspect of initial licensing or oversight of continued operations but to concentrate on the issues that are unique to period of extended operations. *Turkey Point*, CLI 01 17, 54 NRC at 10 (quoting 60 Fed. Reg. at 22,469). Thus, the issue of reanalyzing design basis accidents or analyzing the plant's performance with respect to accidents not within the plant's design basis is not material to the finding the Staff must make. *Turkey Point*, CLI 01 17, 54 NRC at 10 (quoting 60 Fed. Reg. at 22,469).

VI. The Small Portion of the RVI AMP Managed by the Fatigue Monitoring Program is Adequate

Dr. Lahey and New York challenge the Fatigue Monitoring Program more directly in Contention NYS-26.[9] Dr. Lahey asserts that the fatigue monitoring being conducted as part of Entergy's RVI AMP is not sufficiently conservative, and fails to account for the appropriate environmental factors and uncertainty. Lahey Testimony at 62-66; New York SOP at 37-39. A

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[9] The Staff's response to New York's general challenges to the Fatigue Monitoring Program are addressed the Staff's filings related to NYS-26.

careful examination of Dr. Lahey's testimony and New York's assertions shows that the concerns raised regarding the fatigue monitoring program do not challenge RVI components.

The components that Dr. Lahey identifies as being a primary concern are not, in fact, RVI components and are not covered by the RVI AMP. Staff Testimony at Q256-Q257. Thus, these issues are simply not material to whether the RVI AMP adequately manages aging effects for components managed by the program.

Dr. Lahey asserts that the fatigue calculations show components very close to unity are likely to exceed unity when uncertainties are evaluated.[10] As discussed above, the components identified as close to unity are not within the scope of the RVI AMP, and are not material to whether the AMP is adequate. Dr. Lahey has not identified any fatigue calculation as being in error including any additional analysis that may have done for the component. Dr. Lahey, pointedly, does not identify any RVI components as being close to unity such that an error analysis might alter the Staff's conclusions nor does he indicate how the error analysis would alter the conclusions regarding components with significant margin.

[10] As previously explained, these components identified by Dr. Lahey are not part of the RVI AMP.

CONCLUSION

The Staff's conclusions regarding the adequacy of the Applicant's AMP for reactor vessel internals reflect careful evaluation of the Applicant's LRA and related submittals, as presented in the SER and SER Supplement 1, and SER Supplement 2. Further, the Staff's testimony reflects careful consideration of the challenges presented by New York and Dr. Lahey. The Staff's evaluation and conclusions warrant that Contention NYS-25 be resolved in favor of the Applicant.

/Signed (electronically) by/

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