UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

October 24, 2016

Mr. Bryan C. Hanson
President and Chief Nuclear Officer
Exelon Generation Company, LLC
Oyster Creek Nuclear Generating Station
4300 Winfield Road
Warrenville, IL 60555

SUBJECT: R. E. GINNA NUCLEAR POWER PLANT: SAFETY EVALUATION RELATED TO REACTOR VESSEL INTERNALS INSPECTION PLAN BASED ON MRP-227-A (TAC NO. MF6713)

Dear Mr. Hanson:

In a letter dated September 28, 2012, Constellation Energy (the licensee) submitted an aging management program (AMP) for the reactor vessel internals (RVI) at R.E. Ginna Power Plant (Ginna). The Electric Power Research Institute MRP-227-A report, "Pressurized Water Reactor (PWR) Internals Inspection and Evaluation Guidelines," and its supporting reports were used as technical bases for developing Ginna's RVI AMP. The licensee submitted the AMP with the intent to meet the license renewal (LR) Commitment 31 addressed in Appendix A of NUREG-1786, "Safety Evaluation Report Related to the License Renewal of R. E. Ginna Nuclear Power Plant." The AMP included Inspection and Evaluation (I&E) guidelines for the RVI components at Ginna.

The NRC staff has reviewed the inspection plan for the Ginna's RVI components and concludes that the Ginna inspection plan is acceptable because it is consistent with I&E guidelines of MRP-227-A. The licensee addressed all eight applicant/licensee action items and seven conditions specified in MRP-227-A appropriately. Consequently, the licensee has met its license renewal Commitment 31 in Appendix A of NUREG-1786.

If you have any questions, please contact me at (301) 415-2020 or by e-mail at Brenda.Mozafari@nrc.gov.

Sincerely,

Brenda Mozafari

Brenda Mozafari, Senior Project Manager
Plant Licensing Branch I-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-244

cc: Distribution via Listserv

Enclosure:
Safety Evaluation

UNITED STATES
NUCLEAR REGULATORY COMMISSION
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SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

INSPECTION PLAN FOR REACTOR VESSEL INTERNALS

R. E. GINNA NUCLEAR POWER PLANT

CONSTELLATION ENERGY

DOCKET NO.: 50-244

1.0 INTRODUCTION AND BACKGROUND

In a letter dated September 28, 2012 (Reference 1), Constellation Energy (the licensee) submitted an aging management program (AMP) for the reactor vessel internals (RVI) at R.E. Ginna Power Plant (Ginna). The Electric Power Research Institute MRP-227-A report, "Pressurized Water Reactor (PWR) Internals Inspection and Evaluation Guidelines," and its supporting reports were used as technical bases for developing Ginna's RVI AMP (Reference 2). The licensee submitted the AMP with the intent to meet the license renewal (LR) Commitment 31 addressed in Appendix A of NUREG-1786, "Safety Evaluation Report Related to the License Renewal of R. E. Ginna Nuclear Power Plant," (Reference 3). The AMP included Inspection and Evaluation (I&E) guidelines for the RVI components at Ginna.

2.0 REGULATORY EVALUATION

Title 10 of the *Code of Federal Regulations* (CFR) Part 54 (Reference 4) addresses the requirements for plant LR process. The regulation at 10 CFR Section 54.21 requires that each application for LR contain an integrated plant assessment (IPA) and an evaluation of time limited aging analyses (TLAAs). The plant-specific IPA shall identify and list those structures and components subject to an aging management review (AMR) and demonstrate that the effects of aging (e.g., cracking, loss of material, loss of fracture toughness, dimensional changes, and loss of preload) will be adequately managed so that their intended functions will be maintained consistent with the current licensing basis (CLB) for the period of extended operation (PEO) as required by 10 CFR 54.29(a). In addition, 10 CFR 54.22 requires that a license renewal application (LRA) include any technical specification changes or additions necessary to manage the effects of aging during the PEO as part of the LRA.

Structures and components subject to an AMP shall encompass those structures and components that (1) perform an intended function, as described in 10 CFR 54.4, without moving parts or without a change in configuration or properties and (2) are not subject to replacement based on a qualified life or specified time period. These structures and components are referred to as "passive" and "long-lived" structures and components, respectively. The scope of components considered for inspection under MRP-227-A includes core support structures (typically denoted as Examination Category B-N-3 by the American Society of Mechanical Engineers (ASME) Code, Section XI (Reference 5) and those RVI components that serve an intended LR safety function pursuant to criteria in 10 CFR 54.4(a)(1). The scope of the program does not include consumable components such as fuel assemblies, reactivity control assemblies, and nuclear instrumentation because these components are not typically within the scope of the components that are required to be subject to an AMP, as defined by the criteria set in 10 CFR 54.21(a)(1).

The NRC staff's review of the LRA for Ginna was documented in NUREG-1786. The licensee submitted the inspection plan as part of its AMP for the RVI components with the intent to meet its LR Commitment 33 addressed in Appendix A of NUREG-1786 (Reference 6). MRP-227-A summarized most recent industry recommended I&E guidelines for PWR RVI components. The safety evaluation (SE) for MRP-227-A was issued on December 16, 2011 (Reference 7), with seven topical report conditions and eight applicant/licensee action items. The topical report conditions were specified to ensure that certain information was revised generically in the published MRP-227-A, and the applicant/licensee action items were specified for applicant/licensees to address plant-specific issues which could not be resolved generically in the December 16, 2011, SE on MRP-227-A. In fact, almost all actions related to the above mentioned LR Commitments have been accomplished as a result of issuance of the December 16, 2011, SE in MRP-227-A, or are being addressed by the licensee in its plant-specific inspection plan based on MRP-227-A in response to the eight applicant/licensee action items.

3.0 TECHNICAL EVALUATION

The licensee's September 28, 2012, submittal contains Section 1, "Purpose," Section 2, "Background," and Section 3, "Program Owner," Section 6 "Demonstration," Section 7, "Required Program Enhancements/Implementation Schedule," Section 8, "Summary of Implementing Documents/Responsible Department," and Attachments A through G. These Sections contain no specific technical information which would affect the review and approval of the Ginna's inspection plan. Therefore, the focus of the NRC staff's evaluation is related to: Section 4, "Description of Ginna AMP and Industry Program," Section 5, "Ginna's Aging Management Program Attributes," which includes operating experience, and, Section 6, "Demonstration of Applicant/Licensee Action Items and SE Conditions Compliance to SE on MRP-227-A."

3.1 Description of Ginna AMP and Industry Program: Section 4

3.1.1 Baffle-former bolts

Licensee Evaluation

In this section the licensee included its operating experience (OE) related to cracking in baffle-former bolts and at Ginna, these bolts were manufactured with type 347 stainless steel materials. These bolts are susceptible to irradiation-assisted stress corrosion cracking (IASCC), irradiation embrittlement (IE), stress relaxation, and void swelling. To detect the aging effects in these bolts, the licensee used ultrasonic testing (UT) method. During the refueling outage (RFO) in 1999, the licensee replaced fifty six bolts which had reportable indications, and the replaced bolts were made with type 316 stainless steel which is less susceptible to IASSC. The replaced bolts were part of a prequalified minimum bolt pattern for a two-loop Westinghouse unit. Based on the previous OE, in MRP-227-A report, these bolts were categorized under the "Primary" inspection category which requires UT inspections be accessible at every ten year interval. In 2011, the licensee performed UT examinations on the 56 bolts that were previously replaced with type 316 stainless steel which is less susceptible to IASSC. In addition, the licensee conducted UT examinations on 99 baffle former bolts, only one bolt had indications. Based on these results, the licensee concluded that after 25-35 Effective Full Power Operation (EFPY), the failure rate in the baffle former bolts is less than one per cent. Based on the results, the licensee concluded that its corrective actions provides reasonable assurance that the structural integrity of the baffle former bolts is maintained until next inspection.

Staff Evaluation

The staff reviewed the licensee's inspections and replacement program for the baffle former bolts and concludes that the licensees AMP provides reasonable assurance that the baffle former bolts will maintain their functionality until next inspection. The staff's conclusion was based on the following: (1) the licensee replaced bolts with type 316 stainless steel material which has better resistance to IASCC than type 347 material, (2) the replaced bolts have better design than the original ones in resisting cracking, (3) the licensee stated that it complied with the minimum bolting pattern design developed by Westinghouse Owners Group (WOG), (4) the licensee switched to low leakage core design before 30 EFPY which would facilitate reduction in the exposure to neutron fluence in these bolts during the period of extended operation (PEO), (5) combination of reduction in the exposure to neutron fluence, a better bolt design, and superior material will reduce the risk of IASCC in the replaced bolts, and (6) compliance with WOG's minimum bolt pattern criteria would provide adequate assurance that the baffle former bolts will maintain their functionality during the PEO.

3.1.2 Thimble Tubes

Licensee Evaluation

Ginna's_thimble tube inspection program manages cracking due to stress corrosion cracking (SCC) in bottom mounted instrumentation (BMI) guide tube fillet welds and BMI guide tubes. Eddy current testing (ET) is used to detect SCC in the tube assembly. Thimble tube thinning occurs due to wear as a result of flow induced vibration. During RFO in 2011, the licensee stated that it replaced all thirty six flux thimbles with chrome plated tubes in the area from twelve inches above the core plate to twelve inches below the reactor bottom head. The chrome plate is designed to reduce the wear due to flow induced vibration in the thimble tubes. The licensee further stated that it is periodically monitoring the thimble tubes as recommended by the Bulletin 88-09, "Thimble Tube Thinning in Westinghouse Reactors."

Staff Evaluation

The staff reviewed the licensee's evaluation and concludes that the licensee adequately addressed the AMP for the thimble tubes at Ginna, and this conclusion is based on the following: (1) the licensee, based on the OE, had implemented plant-specific inspection and monitoring program to identify the wear in the thimble tubes at Ginna, (2) replacement of all thirty six thimble tubes provides reasonable assurance that the licensee has adequate corrective steps to manage aging degradation in thimble tubes, and, (3) the licensee implemented a plant-specific AMP for thimble tube that is consistent with NUREG 1801, Generic Aging Lessons Learned (GALL)" (Reference 7), AMP XI.M16, "PWR Vessel Internals." GALL AMP addresses various program elements that include guidelines for effective management of the aging degradation in RVI components. Compliance with GALL AMP program elements provides reasonable assurance that the aging degradation (wear) in thimble tubes is adequately managed by the licensee. Therefore, the staff concludes that this issue is adequately addressed by the licensee.

3.2 Ginna Aging Management Program Attributes Evaluation Section 5.0

3.2.1 Aging Management Program Attributes

Licensee Evaluation

In its September 28, 2012, submittal, the licensee stated that the AMP for the RVI components at Ginna is in compliance with all the attributes addressed in NUREG-1801, "Generic Aging Lessons Learned" Report (GALL), Revision 2, Section XI.M16A, "PWR Vessel Internals." The licensee further stated that Ginna fully utilized the GALL process contained in NUREG-1801 in performing aging management review of the RVI components in the license renewal process. Ginna's AMP for the RVI components includes consideration of the augmented inspections identified in MRP-227-A and fully meets the requirements of Section XI.M16A in the GALL Report. The licensee further stated that the augmented inspections, based on program enhancements resulting from industry programs, will become part of the ASME Code, Section XI program.

Staff Evaluation

The staff reviewed the ten program elements for Ginna's AMP and compared them with the program elements XI.M16A in GALL Report, Revision 2, and concluded the following: (1) all ten program elements described in Section 5.0 of the licensee's September 28, 2012, submittal are consistent with the program elements addressed in XI.M16A, in GALL Report, Revision 2, (2) all the program elements included extensive discussions on the implementation of MRP-227-A, I&E guidelines and ASME Code, Section XI ISI criteria, (3) the licensee's discussions included applicability of its Action Items addressed in the staff's SE for MRP-227-A, and (4) the licensee complied with the provisions addressed in MRP-227-A, specifically with respect to the following program elements: (a) parameters monitored or inspected, (b) detection of aging effects, and (c) monitoring and trending.

3.2.2 Control Rod Guide Tube Assembly (Guide Cards)

Licensee Evaluation

The licensee stated that control rod guide cards were binned under "Primary" inspection category in MRP-227-A and that they would be inspected for aging degradation due to wear every 10 years. In Section 5.3.3.2.1 of the submittal dated September 28, 2012, the licensee stated that it inspected twenty nine control rod guide tube (CRGT) assemblies which contained 261 guide cards (9 guide cards per one CRGT) that were accessible during the RFO 2011, and found low amount of wear. Based on this observation and generic evaluation developed by Westinghouse (the licensee), concluded that any significant amount of wear is unlikely to occur in the guide cards during the first period of the PEO. The licensee stated that it will continue to inspect guide cards in accordance with I&E guidelines of MRP-227-A during the PEO.

Staff Evaluation

Based on the licensee's response, the staff concludes that the AMP for guide cards is adequately managed at Ginna. The staff's conclusion is based on the following. (1) no aggressive aging degradation due to wear was observed at Ginna, (2) guide cards are inspected every ten year interval in accordance with I&E guidelines addressed in MRP-227-A, (3) the licensee would use the feet-wide inspection results and the generic evaluation developed by Westinghouse in establishing the subsequent inspection frequency and, (4) if any unacceptable wear in guide cards were to be observed during the future inspections, the licensee, as part of its corrective action, would determine subsequent inspection frequency for the guide cards based on the extent of wear observed in these cards. Based on the evaluation stated above, the staff concludes that the licensee adequately demonstrated that its AMP is effective in identifying wear in guide cards in a timely manner. Therefore, the staff considers that the issue is adequately addressed by the licensee.

3.2.3 Materials Susceptible to Degradation

OE in the PWR fleet to date identified that nickel base and stainless steel alloys are susceptible to some of the aging degradation mechanisms addressed in MRP-227-A. In this context, by a letter dated January 7, 2016 (Reference 8), in RAI-2, NRC staff requested that the licensee confirm that the following materials are not currently used in the RVI components binned under categories addressed in MRP-227-A: nickel base alloys (i.e., inconel 600), weld metals (i.e., Alloy 82 and 182), Alloy X-750 (excluding control rod guide tube split pins), Alloy A-286, ASTM A 453 (Grade 660, Condition A or B), type stainless steel 347 material (excluding baffle-former and barrel former bolts), precipitation hardened (PH) stainless steel materials (i.e.,17-4 and 15-5) and type 431 stainless steel material.

Licensee's Evaluation

In its response to RAI-2, dated February 11, 2016 (Reference 9), the licensee stated that the following materials are not used at Ginna: Alloy A-286, ASTM A 453 (Grade 660, Condition A or B), PH stainless steel materials (17-4 and 15-5), type 347 stainless steel materials (excluding baffle-former and barrel former bolts), and type 431 stainless steel material.

However, the licensee identified the following material which was used in RVI components at Ginna: Alloy X-750 is used in clevis insert bolts and this material is prone to primary water stress corrosion cracking (PWSCC).

Staff Evaluation

The staff reviewed the response and based on the information provided, the staff determined the following: (1) baffle-former bolts and barrel former bolts are binned under "Primary," and "Expansion" inspection categories respectively, and their aging degradation is monitored using I&E guidelines of MRP-227-A, (2) clevis inserts are routinely inspected in accordance with the requirements of ASME Code, Section XI. The licensee, in an e-mail correspondence dated March 17, 2016 (Reference 10), stated that visual examination of clevis insert bolts that was performed during RFO 2011, showed no cracks. Based on the response, the staff concludes that aging degradation in clevis insert bolts described above are adequately monitored using routine inspections as required by ASME Code, Section XI criteria, and by the criteria addressed in I&E guidelines of MRP-227-A. Therefore, the staff considers that the issue addressed in RAI-2 is closed.

3.3 Applicant/Licensee Action Item (AI) 1 of SE for MRP-227-A

3.3.1 Evaluation of the Licensee's Resolution of AI 1

Section 4.2.1 of the SE for MRP-227-A states that "Each applicant/licensee should refer, in particular, to the assumptions regarding plant design and operating history made in the failure modes, effects and criticality analysis (FMECA) and functionality analyses for reactors of their design (i.e., Westinghouse, Combustion Engineering (CE), or Babcock and Wilcox (B&W) which support MRP-227. The applicant/licensee shall submit this evaluation for NRC review and approval as part of its application to implement the approved version of MRP-227. This is Applicant/Licensee Action Item 1."

To resolve the generic issue of the information needed from licensees to resolve Action Item 1, a series of proprietary and public meetings were conducted, at which the NRC, Westinghouse, the Electric Power Research Institute (EPRI), and utility representatives discussed regulatory concerns and determined a path for a comprehensive and consistent utility response to demonstrate applicability of MRP-227-A, specifically for Westinghouse and CE-design PWR RVI. A summary of the proprietary meeting presentations and supporting proprietary generic design bases information are contained in Westinghouse proprietary report WCAP-17780-P, "Reactor Internals Aging Management MRP-227-A Applicability for Combustion Engineering and Westinghouse Pressurized Water Reactor Designs," (Reference 11). This report provides background on the proprietary design information regarding variances in stress, fluence, and temperature in the RVI components that were designed by Westinghouse and CE to support NRC reviews of utility submittals to demonstrate plant-specific applicability of MRP-227-A.

As a result of the technical discussions with the NRC staff, the basis for a plant to respond to the NRC's RAI to demonstrate compliance with MRP-227-A for originally licensed and uprated conditions was determined to be satisfied with plant-specific responses to the following two generic questions:

Question 1: Does the plant have non-weld or bolting austenitic stainless steel components with 20 percent cold work or greater, and, if so, do the affected components have operating stresses greater than 30 ksi?

(If both conditions are true, additional components may need to be screened in for stress corrosion cracking, SCC.)

Question 2: Does the plant have atypical fuel design or fuel management that could render the assumptions of MRP-227-A, regarding core loading/core design, non-representative for that plant?

By MRP Letter 2013-025 dated October 14, 2013, EPRI provided to licensees "MRP-227-A Applicability Guidelines for Combustion Engineering and Westinghouse Pressurized Water Reactor Designs, (MRP-227-A Applicability Guidelines), a non-proprietary document (Reference 12) containing guidance for responding to the two questions above. The staff assessed MRP Letter 2013-025 and the technical basis contained in WCAP-17780-P, and concludes that if an applicant or licensee demonstrates that its plant(s) comply with the guidance in MRP Letter 2013-025, there is reasonable assurance that the I&E guidance of

MRP-227-A will be applicable to the specific plant(s). The details of the staff's non-proprietary assessment of WCAP-17780-P and MRP Letter 2013-025 in Reference 13. The guidance in MRP Letter 2013-025 provides an acceptable basis for licensees to respond to the generic questions 1 and 2 addressed above.

In a letter dated January 7, 2016 (Reference 8), in RAI-1, the staff requested that the licensee provide the information (discussed in References 1 and 2), related to verification of the applicability of MRP-227- to Ginna.

With respect to Question 1, the applicability guidelines addressed in MRP Letter 2013-025 provides guidance for the licensees to assess whether RVI components at their plant other than those identified in the generic evaluation, have the potential for cold work greater than twenty percent. In response to Question 1, by letter dated February 11, 2016 (Reference 9), the licensee stated that it intends to comply with the generic guidance developed by Pressurized Water Reactor Owners Group (PWROG), and that this guidance will be submitted to the staff for information only. By letter dated June 15, 2016 (Reference 14), the PWROG submitted a generic guidance, PWROG-15105-NP, Revision 0, PWR RV Internals Clod Work Assessment." (Reference 15), which addressed the effect of cold work on the occurrence of SCC in RVI components. The NRC staff's assessment of this report entailed an evaluation of the applicability of the generic guidance to resolve the issue related to the effect of cold work on SCC in the RVI components at Ginna. The staff noted that to date the PWROG evaluated the effects of cold work on SCC of RVI components in fifty six percent of Westinghouse designed units. The generic evaluation of the RVI components entailed a detailed review of the following: (1) material specification requirements; (2) cold work induced by fabrication; (3) review of the component drawings; (4) heat treatment; (5) limitation on the maximum allowed strength and hardness values addressed in material specifications or component drawings.

Based on its assessment of PWROG-15105-NP, Revision 0, the staff noted the following: Extensive research on component drawings and material specifications of all RVI components in fifty percent of the Westinghouse fleet were performed by PWROG. Component drawings and material specifications for the RVI components included the following requirements: (a) type of material; (b) solution annealing heat treatment; and (c) cold work after solution annealing. Based on the specifications and component drawings, the PWROG identified potential for cold work in the RVI components which resulted in binning RVI components in the following five categories included in MRP Letter 2013-025: Category 1- cast austenitic stainless steel (CASS); (2) Category 2- hot formed stainless steel; (3) Category 3-annealed austenitic stainless steels; (4) Category 4-fasteners austenitic stainless steels; and, (5) Category 5-cold formed austenitic stainless steels without subsequent solution annealing.

Materials binned under Categories 1, 2 and 3 contain no greater than twenty percent of cold work due to controlled fabrication and compliance with material specifications. Therefore, the RVI components binned under Categories 1, 2 and 3 are consistent with MRP Letter 2013-025 guidelines. Only materials that fall under Categories 4 and 5 were treated as cold worked and they were evaluated as such. For components binned under Category 4, cold work greater than twenty percent was already considered and these materials are fasteners in the Westinghouse's generic aging evaluation of the RVI components. These materials were conservatively assumed to be susceptible to SCC and they were included in MRP-227-A. This generic evaluation is addressed in MRP-191 Revision 0, "Materials Reliability Program: Screening, Categorization and Ranking of Reactor Internals of Westinghouse and Combustion Engineering PWR Designs" (Reference 16). Since MRP-191 was used as a basis for establishing I&E

guidelines in MRP-227-A, it was concluded that no additional stress analyses are required for Category 4 RVI components.

For Category 5 solution annealed components followed by cold work that could potentially exceed twenty percent, further evaluation by PWROG included a review of fleet wide material specification and component drawings designed by Westinghouse. Based on the review, it was concluded that the allowed cold work was limited by the maximum limit on tensile strength or hardness that was imposed by material specification or component drawings. Fabrication induced cold work on RVI components, such as bending, was limited to maximum cold work of three percent. Therefore, it was concluded that Category 5 RVI components in Westinghouse units were not subject to cold work greater than twenty percent, hence they are not susceptible to SCC.

The licensee, in Attachment H (Action Item 2) of its September 28, 2012, submittal, stated that it performed the scoping and screening of the RVI components as per the requirements of the license renewal process. RVI materials used at Ginna are consistent with the materials specified in Table 4-4 of MRP-191 which was used as a technical basis document for the development of I&E guidelines in MRP-227-A. The staff noted that the materials listed in Table 4-4 of MRP-191 are consistent with the materials listed in Table C-1 of the generic report PWROG-15105. Therefore, the staff concluded that the generic evaluation of the effect of cold work on SCC that is addressed in PWROG-15105 would be applicable to Ginna.

Based on its assessment of PWROG-15105, the staff accepts the applicability of PWROG's generic evaluation on the effect of cold work on SCC at Ginna's RVI components for the following reasons:

(1) The staff noted that the RVI components used in units designed by Westinghouse were not subject to cold work greater than twenty percent and this conclusion was substantiated by the limitations prescribed in material specification and component drawing,

(2) PWROG performed fleet wide review of the RVI components and this review entailed identification of: (a) component drawing, (b) type of the material and material specification, (c) solution annealing heat treatment, and (d) cold work after solution annealing during fabrication. Majority of the RVI components contain no greater than twenty percent of cold work due to controlled fabrication and compliance with material specifications. These components are not susceptible to SCC. (Categories 1, 2 and 3),

(3) Fasteners were exposed to cold work greater than twenty percent, and therefore, they were already included in the review of MRP-227-A. (Category 4), and,

(4) RVI components that are exposed to cold work greater than twenty percent are most susceptible to SCC. PWROG's generic review revealed that the allowed cold work during initial fabrication for these components was limited by the maximum limit on the tensile strength or hardness that was imposed by material specification or component drawings. Therefore, based on its evaluation, PWROG concluded that these components were not subject to cold work greater than twenty percent. (Category 5).

Based on the above evaluation the staff concluded that the licensee complied with the guidelines provided in MRP Letter 2013-025 for RVI components that were binned under Categories 1 through 5. The staff also noted that the RVI components binned under Categories 1 through 5 were not exposed to cold work greater than twenty percent. Therefore, the staff concludes that the licensee has demonstrated that it adequately evaluated the effects cold work on SCC in RVI components binned under Categories 1 through 5 at Ginna. Therefore, the staff concludes that the licensee complied with the guidelines (with respect to the evaluation of cold work on SCC issue) addressed in MRP Letter 2013-025.

With respect to Question 2, the MRP Letter 2013-025 provide quantitative criteria to allow a licensee to assess whether a particular plant has atypical fuel design or fuel management. For a Westinghouse design plant such as Ginna, these criteria are:

(1) The heat generation rate must be ≤ 68 Watts/cm^3.

(2) The maximum average core power density must be less than 124 Watts/cm^3.

(3) The active fuel to upper core plate (UCP) distance must be greater than 12.2 inches.

In response to RAI-(2), in a letter dated February 11, 2016 (Reference 9), the licensee stated that to date, it complied with the criteria stated above for the essential attributes related to maximum average core power density and active fuel distance to UCP. However, the licensee stated that Ginna's heat generation rates for a few reload cycles did not comply with the criteria (maximum 68 Watts/cm^3) addressed in MRP Letter 2013-025. These reload cycles were: 1B, 2, 4, 5, 6, 7, 8, 9, 10, 11, 12. The licensee evaluated the noncompliance of the heat generation rates and concluded the following. Criteria in MRP Letter 2013-025 allows a deviation for a period of two years, however, in Ginna's case, the deviation exceeded this value by 5.3 EFPY. The licensee stated that the duration period in which the deviation occurred was offset by many years of operation (since 1983) where the heat generation rate values were within the allowed limit (maximum 68 Watts/cm^3). The licensee stated that it will continue to operate Ginna unit within the allowed limits during PEO.

The staff reviewed the licensee's evaluation and concludes the following: I&E guidelines in MRP-227-A were developed with the assumption that the unit is expected to operate using out-in core fuel configuration for the first thirty years of operation which results in high neutron leakage during the first thirty years of operation followed by low leakage core for the remainder life of the unit. This assumption is very conservative because high neutron exposure during the first thirty years of operation results in high level of damage on the RVI components. However, majority of the Westinghouse units switched to low leakage earlier than thirty years of operation. Therefore, RVI components are exposed to lower neutron fluence, which is still below the damaging level of fast neutron fluence value used in developing MRP-227-A. Ginna operated within the allowable limits of the criteria addressed in MRP Letter 2013-025 since 1983. Therefore, Ginna, is still bounded for the RVI components receiving less fast neutron fluence than addressed in MRP Letter 2013-025. Based on the submitted response, the staff concludes that the licensee satisfied the guidelines related to the fuel management issue addressed in MRP Letter 2013-025 and that implementation of I&E guidelines addressed in MRP-227-A would be valid during the PEO. Based on this review, the staff considers that the licensee addressed AI 1 satisfactorily.

3.3.2 Evaluation of the Licensee's Resolution of AI 2

Section 4.2.2 of the SE for MRP-227-A states that "each applicant/licensee is responsible for identifying which RVI components are within the scope of LR for its facility. Applicants/licensees shall review the information in Tables 4-4 and 4-5 in MRP-191 and identify whether these tables contain all of the RVI components that are within the scope of LR for their facilities in accordance with 10 CFR 54.4. If the tables do not identify all the RVI components that are within the scope of LR for its facility, the applicant or licensee shall identify the missing component(s) and propose any necessary modifications to the program defined in MRP-227, as modified by this SE, when submitting its plant-specific AMP such that the effects of aging on the missing component(s) will be managed for the PEO."

Licensee Evaluation

The licensee, in its September 28, 2012, submittal stated that it performed the scoping and screening of the RVI components as per the requirements of the license renewal process. RVI materials used at Ginna are consistent with the materials specified in Table 4-4 of MRP-191 which was used as a technical basis document for the development of I&E guidelines in MRP-227-A.

Staff Evaluation

The staff reviewed the licensee's evaluation and concludes that: (1) the licensee's AMP for the RVI components is consistent with MRP-227-A I&E guidelines; (2) no additional RVI components at Ginna were screened in due to the usage of different type of materials that were not prescribed in MRP-191/MRP-227-A; and, (3) the licensee complied with the guidelines addressed in Action Item 1. Details of the staff's evaluation of the Action Item 1 are addressed in Section 3.4.1 of this SE. Based on this assessment, the staff considers that the licensee addressed the Action Item 2 satisfactorily.

3.3.3 Evaluation of the Licensee's Resolution of AI 3

Section 4.2.3 of the SE for MRP-227-A states that "applicants/licensees of Westinghouse are required to perform plant-specific analysis either to justify the acceptability of an applicant's/licensee's existing programs, or to identify changes to the programs that should be implemented to manage the aging of these components for the PEO. The results of this plant-specific analyses and a description of the plant-specific programs being relied on to manage aging of these components shall be submitted as part of the applicant's/licensee's AMP application. The Westinghouse components identified for this type of plant-specific evaluation include: Westinghouse guide tube support pins (split pins)."

Licensee Evaluation

The licensee stated that CRGT split pins at Ginna were fabricated from Alloy X-750 which did not receive high temperature heat treatment (HTH). Alloy X-750 material without undergoing HTH treatment is more susceptible to PWSCC. The licensee, in Section 4.2.1 of the September 28, 2012, submittal, stated that as part of its corrective action during the 2011 refueling outage, replaced Alloy X-750 with cold worked type 316 stainless steel material which has superior resistance to PWSCC. Therefore, the licensee determined that no further inspections of CRGT split pins is required for split pins.

Staff Evaluation

The staff reviewed the licensee's evaluation and concludes that the replaced CRGT split pins would be more resistant to PWSCC, hence, cracking is unlikely in these split pins. The staff determined that replacement with cold worked type 316 material provides reasonable assurance that the aging degradation in this component is adequately monitored by the licensee during the PEO. Based on the licensee's evaluation, the staff concludes that the issue related to Action Item 3 of the staff's SE for the MRP-227-A is resolved.

3.3.4 Evaluation of the Licensee's Resolution of AIs 4 and 6

Action Items 4 and 6 of the staff's SE for the MRP-227-A report are applicable to the RVI components designed by B&W, and therefore, they are not applicable to Ginna.

3.3.5 Evaluation of the Licensee's Resolution of AI 5

Section 4.2.5 of the SE for MRP-227-A states that "applicants/licensees shall identify plant-specific acceptance criteria to be applied when performing the physical measurements required by the NRC-approved version of MRP-227 for loss of compressibility for Westinghouse hold down springs. The applicant/licensee shall include its proposed acceptance criteria and an explanation of how the proposed acceptance criteria are consistent with the plants' licensing basis and the need to maintain the functionality of the component being inspected under all licensing basis conditions of operation as part of their submittal to apply the approved version of MRP-227."

Action Item 5 in the staff's SE for MRP-227-A states that the licensee should identify a plant-specific acceptance criterion to be applied while performing the physical measurement of loss of compressibility for hold-down spring. Hold down springs are fabricated with 304 austenitic stainless steel material which is susceptible to loss of preload due to irradiated assisted creep/stress relaxation.

Licensee's Evaluation

The licensee stated that the hold down spring at Ginna is 403 stainless steel material and therefore, the licensee determined that AI 5 did not apply to Ginna.

Staff Evaluation

The staff noted that AI 5 is applicable to 304 stainless steel material only, and 403 stainless steel has more resistant to creep/stress relaxation than 304 stainless steel material. This is due to the fact that the stiffness value of 403 stainless steel material is higher than that of 304 stainless steel material. Therefore, the staff concluded that the replacement with 403 stainless steel hold-down spring provides reasonable assurance that the AMP for this item is adequately managed during the PEO, and the issue related to AI 5 of the staff's SE for the MRP-227-A is resolved.

3.3.6 Evaluation of the Licensee's Resolution of AI 7

Action Item 7 was discussed in Section 3.3.7 of the staff's SE for the MRP-227-A, and it requires that the licensees of Westinghouse reactors are required to develop plant-specific analyses to be applied for their facilities to demonstrate that lower support columns (LSCs) cast austenitic stainless steel (CASS) will maintain their function during the PEO. This component is subject to irradiation embrittlement (IE) and thermal embrittlement (TE). CASS materials with delta ferrite greater than twenty percent would be susceptible to loss of fracture toughness due to TE. CASS materials that are exposed to neutron fluence value greater than $1x10^{17}$ n/cm^2 (E>1 MeV) are susceptible to IE.

Licensee Evaluation

In Attachment H in its September 28, 2012, submittal, the licensee stated that LSCs at Ginna were fabricated with wrought austenitic stainless steel (type 304) material. Therefore, this material which is a wrought product, is not susceptible to TE.

Staff Evaluation

The staff noted that the IE is still an active aging degradation in the LSCs. In Table 4-6, of MRP-227-A, LSCs were binned under "Expansion Inspection" category and the corresponding Primary link is upper core barrel flange weld which is a Primary indicator SCC. The staff noted that primary indicator of the aging effect due to IE and irradiation assisted stress corrosion cracking (IASCC) would be a RVI component (binned under "Primary" inspection category) that is susceptible to these aging degradation mechanisms. Therefore, in order to identify aging effects due to IE and IASCC, a suitable Primary link for LSCs is to be selected. Any cracking due to IE and IASCC would be identified in this Primary link RVI component much sooner than in the LSCs.

The staff, in its SE dated December 18, 2015, had established a precedent setting inspection guideline for LSCs at Turkey Point Nuclear Generating Unit Nos. 3 and 4 (Reference 17). Based on a comparison of neutron fluence, the staff determined that the lower core barrel girth weld would be a more appropriate predictor of IE. Additionally, the lower core barrel girth welds selected as Primary link would be a leading indicator of IASCC because welds generally contain residual tensile stresses, whereas the LSCs would likely be in compressive stress under normal operating conditions and thus should be much less susceptible to IASCC. Therefore, the staff requests that the licensee revise Table 1.2 in Attachment E of the September 28, 2012, submittal.

Based on the aforementioned evaluation and pending the revision of the licensee's I&E guidelines addressed above, the staff concludes that the issue related to AI 7 of the staff's SE for the MRP-227-A is resolved.

3.3.7 Evaluation of the Licensee's Resolution of AI 8

The Action Item 8 of the staff's SE for the MRP-227-A requires that the licensee submit the AMP for the RVI components that is consistent with I&E guidelines addressed in MRP-227-A. In its submittal dated September 28, 2012, the licensee included AMP and I&E guidelines for RVI components that are consistent with the MRP-227-A guidelines. The staff determined that the licensee is in compliance with the Action Item 8 and therefore, it considers that this issue is closed.

3.4 TR Conditions in the Staff's SE for the MRP-227-A

The staff SE for MRP-227 contains seven conditions that the licensee must follow to receive credit for MRP-227-A implementation. The NRC staff reviewed the licensee's submittal against these seven conditions.

Condition 1: The licensee, in its inspection program addressed in Table 1.2 in Attachment E of the September 28, 2012, submittal has added the upper core plate and lower support forging or casting to its RVI inspection program. This addition is consistent with the guidelines addressed in Table 4-6 of the MRP-227-A report; therefore, the staff finds Condition 1 is met.

Condition 2: Consistent with the I&E guidelines addressed in Table 4-3 of the MRP-227-A report, the licensee included the upper and lower core barrel welds and lower core barrel flange in its AMP. Therefore, the staff finds Condition 2 is met.

Condition 3: Condition 3 is not applicable to Westinghouse designed RVI components, and therefore, the staff the staff finds Condition 3 is met.

Condition 4: A criterion for a minimum area of inspection coverage is addressed in this condition. This criterion states that a minimum of 75% coverage of the entire examination volume (i.e., including both accessible and inaccessible regions) of the RVI components and their welds, and a minimum sample size of 75% of the total population of like components (e.g., bolts) should be inspected. The licensee included this guideline in its AMP; therefore, the staff finds Condition 4 is met.

Condition 5: This condition states that a 10-year inspection frequency for baffle-former bolts in Westinghouse-designed reactors should be implemented following the initial or baseline inspection. The licensee satisfied this condition by including this criterion in its Table 1.1 in Attachment E of the September 28, 2012, submittal; therefore, the staff finds Condition 5 is met.

Condition 6: Condition 6 states that subsequent re-examination for all "Expansion" inspection category components should be at a 10 year interval once degradation is identified in the associated "Primary" inspection category component. The licensee included this guideline in Table 1.2 in Attachment E of the September 28, 2012, submittal; therefore, the NRC staff finds Condition 6 is met.

Condition 7: In Section 5.0 of the September 26, 2012, submittal, the licensee stated that the operating experience related to the aging degradation of the RVI components in the PWR fleet would be periodically documented. Furthermore, the licensee included the operating experience related to the aging degradation of some of the RVI components at Ginna. The staff's review of the operating experience at Ginna is addressed in Sections 3. and 3.2 of this SE. Based on the review, the staff found that the licensee provided the necessary information required by MRP-227-A. The staff finds Condition 7 is met.

Based on the review of the licensee's responses to the seven conditions, the NRC staff concludes that the licensee had adequately addressed all the conditions stated in the NRC staff's SE for MRP-227-A.

4.0 CONCLUSION

The NRC staff has reviewed the inspection plan for the Ginna's RVI components and concludes that the Ginna inspection plan is acceptable because it is consistent with I&E guidelines of MRP-227-A. The licensee addressed all eight applicant/licensee action items and seven conditions specified in MRP-227-A appropriately. Consequently, the licensee has met its license renewal Commitment 31 in Appendix A of NUREG-1786.

5.0 REFERENCES:

1. Letter to NRC Document Control Desk transmitting "License Renewal Aging Management Submit Revised Reactor Vessel Internals Program Document in Accordance with RIS 2011-07," for R. E. Ginna Nuclear Power Plant, September 28, 2012, Agencywide Documents Access and Management System (ADAMS) Accession No. ML12277A174.

2. Electric Power Research Institute (EPRI) "Transmittal: PWR Reactor Internals Inspection Evaluation Guidelines (MRP-227-A), January 9, 2012 (ADAMS Accession Nos. ML12017A193 to ML12017A199).

3. USNRC, NUREG-1786, "Safety Evaluation Report Related to the License Renewal of R. E. Ginna Nuclear Power Plant," May 31, 2004 (ADAMS Accession No. ML041400502).

4. Federal Regulation, Title 10 of the *U.S. Code of Federal Regulations*,(10 CFR) Part 54 "Requirements for Renewal of Operating Licenses for Nuclear Power Plants."

5. American Society of Mechanical Engineers (ASME) Code, Section XI, Regulatory Guide 1.147, Revision 17 "Inservice Inspection Code Case Acceptability, ASME Section XI, Division 1, August 2014 (ADAMS Accession ML13339A689).

6. USNRC, Letter From Robert A. Nelson, Deputy Director, Division of Policy and Rulemaking to Neil Wilmshurst, Vice President, Electric Power Research Institute, Re: Revision 1 to the Final Safety Evaluation of Electric Power Research Institute (EPRI) Report, Materials Reliability Program (MRP) Report 1016596 (MRP-227), Revision 0, "Pressurized Water Reactor (PWR) Internals Inspection and Evaluation Guidelines" December 16, 2011 (ADAMS Accession No. ML11308A770).

7. USNRC, NUREG-1801, Rev. 2 "Generic Aging Lessons Learned (GALL) Report" Final Report, December 31, 2010 (ADAMS Accession No. ML103490041).

8. USNRC, Letter From Diane Render, Project Manager to Bryan C. Hanson, President and CE), Exelon Nuclear, Re: R. E. Ginna Nuclear Power Plant – Request for Additional Information Regarding Reactor Vessels Internals Program" – January 7, 2016 (ADAMS Accession No. ML15349B025).

9. Letter, From James Bastow, Exelon Generation Company, LLC to NRC Document Control Desk, R. E. Ginna Nuclear Power Plant, Renewed Facility Operating License, Re: Response to Request for Additional Information Regarding Reactor Vessel Internals Program," February 11, 2016 (ADAMS Accession No. ML16042A421).

10. USNRC Email, From Laura Lynch, Exelon Nuclear to Diane Render, Project Manager, NRC, Re: Ginna Reactor Vessel Internals Program Document," March 17, 2016 (ADAMS Accession No. ML16077A108).

11. Westinghouse Report, WCAP-17780-P, Rev. 0, "Reactor Internals Aging Management MRP-227-A Applicability for Combustion Engineering and Westinghouse Pressurized Water Reactor Designs," June 2013 (ADAMS Accession No. ML13183A373).

12. Electric Power Research Institute (EPRI) "MRP-227-A Applicability Template Guideline - MRP-227-A Applicability Guidelines for Combustion Engineering and Westinghouse Pressurized Water Reactor Designs," October 14, 2013 (ADAMS Accession No. ML13322A454).

13. USNRC, Office of Nuclear Reactor Regulation – Evaluation of WCAP-17780-P, "Reactor Internals Aging Management MRP-227-A Applicability for Combustion Engineering (CE) and Westinghouse Electric Company (Westinghouse) Pressurized Water Reactor Designs, and MRP-227-A, Applicability Guidelines for CE and Westinghouse Pressurized Water Reactor Designs," November 7, 2014 (ADAMS Accession No. 14309A484).

14. Letter from Jack Stringfellow, SNOC, Pressurized Water Reactor (PWR) Owners Group to USNRC, Re: PWR Owners Group – Submittal of PWROG-15105-NP, Revision 0, "PWR RV Internals Cold-Work Assessment" to the NRC for Information Only (PA-MSC-1288)," June 15, 2016 (ADAMS Accession No. ML16222A299).

15. PWROG-15105-NP, Revision 0, "PA-MSC-1288 - PWR RV Internals Cold-Work Assessment," April 2016 (ADAMS Accession No. ML16222A300).

16. Electric Power Research Institute (EPRI) "Materials Reliability Program: Screening, Categorization, and Ranking of Reactor Internals Components for Westinghouse and Combustion Engineering PWR Design (MRP-191), November 30, 2006 (ADAMS Accession No. ML091910130).

17. USNRC Letter from Benjamin G. Beasley, Chief to Mano Nazar, President and CNO of NextEra Energy, Re: Turkey Point Nuclear Generating Unit Nos. 3 and 4 – Staff Assessment of License Renewal Commitment for Reactor Vessel Internals Implementation Report and Inspection Plan," December 18, 2015 (ADAMS Accession No. ML15336A046).

If you have any questions, please contact me at (301) 415-2020 or by e-mail at Brenda.Mozafari@nrc.gov.

Sincerely,

/RA/

Brenda Mozafari, Senior Project Manager
Plant Licensing Branch I-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

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