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

October 16, 2012

Mr. Joseph E. Pacher
Vice President R.E. Ginna Nuclear Power Plant
R.E. Ginna Nuclear Power Plant, LLC
1503 Lake Road
Ontario, NY 14519

SUBJECT: R.E. GINNA NUCLEAR POWER PLANT – RE: RELIEF REQUEST NO. ISI-09 AUTHORIZATION OF ALTERNATIVE FOR COMPONENT COOLING WATER 1 INCH HALF COUPLING WELD LEAK (TAC NO. ME9519)

Dear Mr. Pacher:

By letters dated September 6, 2012 and September 18, 2012, Constellation Energy Nuclear Group (CENG, the licensee) requested the U.S. Nuclear Regulatory Commission (NRC, Commission) for relief from the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (B&PV Code), Section XI, 2004 Edition, No Addenda. In Relief Request ISI-09, the licensee asked for relief from the ASME Code, Section XI, Mandatory Appendix IX requirement to perform the repairs in accordance with the ASME Code, Section XI, during the next scheduled refueling outage (RFO) at R. E. Ginna Nuclear Power Plant.

As set forth in the attached safety evaluation, the NRC staff has evaluated the licensee's request and concluded that the proposed alternative provides reasonable assurance of structural integrity and leak tightness of the leaking temperature element TE-621. The NRC staff also concludes that complying with the specified ASME Code requirement would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety. Accordingly, the NRC staff concludes that the licensee has adequately addressed all of the regulatory requirements set forth in 10 CFR 50.55a(a)(3)(ii) and is in compliance with the requirements of the ASME Code, Section XI.

Therefore, the NRC staff authorizes the licensee's Relief Request Number ISI-09 at the R.E. Ginna Nuclear Power Plant until a permanent repair can be implemented during the 2014 RFO, Cycle Number 38, scheduled for May 2014.

All other ASME Code, Section XI requirements for which relief has not been specifically requested and approved in this relief request remain applicable, including third party review by the Authorized Nuclear Inservice Inspector.

If you have any questions, please contact the R.E. Ginna Project Manager, Mohan Thadani, at (301) 415-1476.

Sincerely,

George Wilson, Chief
Plant Licensing Branch I-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-244

Enclosure:
As stated

cc w/encl: Distribution via Listserv

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

RELIEF REQUEST NO. ISI-09, RELIEF FROM

AMERICAN SOCIETY OF MECHANICAL ENGINEERS CODE, SECTION XI

FOR COMPONENT COOLING WATER HALF COUPLING WELD LEAK

R.E. GINNA NUCLEAR POWER PLANT, LLC

R.E. GINNA NUCLEAR POWER PLANT

DOCKET NO. 50-244

1.0 INTRODUCTION

By letters dated September 6, 2012 (Agencywide Documents Access and Management System (ADAMS) Accession Number ML12254A378) and September 18, 2012 (ADAMS Accession Number ML12264A591), Constellation Energy Nuclear Group (CENG, the licensee) requested the U.S. Nuclear Regulatory Commission (NRC, Commission) for relief from the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (B&PV Code), Section XI, 2004 Edition, No Addenda, for Component Cooling Water (CCW) 1 inch half coupling weld leak.

In Relief Request ISI-09, the licensee asked for relief from the ASME Code, Section XI, Mandatory Appendix IX requirement to perform the repairs in accordance with the ASME B&PV Code, Section XI, during the next scheduled refueling outage (RFO).

2.0 REGULATORY EVALUATION

Pursuant to 10 CFR 50.55a(g)(4), ASME Code Class 1, 2, and 3 components (including supports) will meet the requirements, except the design and access provisions and the preservice examination requirements, set forth in the ASME Code, Section XI, "Rules for Inservice Inspection (ISI) of Nuclear Power Plant Components," to the extent practical within the limitations of design, geometry, and materials of construction of the components. The regulations require that inservice examination of components and system pressure tests conducted during the first 10-year ISI interval and subsequent intervals comply with the requirements in the latest edition and addenda of Section XI of the ASME Code incorporated by reference in 10 CFR 50.55a(b) 12 months prior to the start of the 120-month interval, subject to the conditions listed therein.

Pursuant to 10 CFR 50.55a(a)(3), alternatives to the ASME Code requirements may be authorized by the NRC if the licensee demonstrates that: (i) the proposed alternative provides an acceptable level of quality and safety, or (ii) compliance with the specified requirements

would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety.

Based on the above, and subject to the following technical evaluation, the NRC staff finds that regulatory authority exists for the licensee to request and the Commission to authorize the alternative requested by the licensee.

3.0 TECHNICAL EVALUATION

3.1. ASME Code Component(s) Affected

Class 3, Component Cooling Water System, 1-inch half-coupling fillet weld associated with Temperature Element TE-621.

3.2 Applicable Code Edition and Addenda

ASME Code, Section XI, 2004 Edition, No Addenda.

3.3 Applicable Code Requirement

The ASME Code, Section XI, 2004 Edition, Paragraph IWA-4133 specifies that mechanical clamping devices used to replace piping pressure boundary shall meet the requirements of Mandatory Appendix IX.

3.4 Reason for Request

On July 11, 2012, the licensee detected a small leak of 1 drop per 4 minutes on a 1-inch half-coupling to fillet weld associated with temperature element TE-621 off a 14-inch Class 3, CCW pipe. The temperature element is located on a common pipe run downstream of both CCW heat exchangers. The degraded fillet weld is the second weld out from the temperature element branch connection off of the 14-inch common header pipe.

The licensee performed a VT-1 visual examination of the leaking location in accordance with the requirements of the ASME Code, Section XI. The licensee characterized the defect as an original welding imperfection, near a weld stop/start, with some minor undercut, along with lack of fusion where the fillet weld fuses to the half-coupling, approximately 3/8 inch to 1/2 inch long. The location of the thermal well is approximately 1 inch above the sleeve for pipe support CCU-138, where it passes through the floor penetration.

The licensee noted that the proximity to the pipe support sleeve makes this area difficult to access for welding. The licensee believes that the restricted access for the welder is potentially a contributing cause of the lack of fusion. The licensee examined the thickness of the base metal on both sides of the weld prior to installing the mechanical clamp and did not detect material loss. In the September 18, 2012, letter, the licensee stated that a supplemental radiographic test performed on September 12, 2012, confirmed that the defect was caused by the lack of fusion.

The system fluid is potassium chromated water at an operating temperature of approximately 90 degrees F and an operating pressure of approximately 85 psig. The system design temperature and pressure at the location is 200 degrees F and 150 psig, respectively. The

licensee stated that as a result of the location of this leak, the degraded weld cannot be isolated from the common header for an ASME Code repair without shutting down all CCW operations. Therefore, to perform the ASME Code repair, the licensee would need to complete a core defuel to establish a plant condition when CCW is not required. The upcoming 2012 RFO will start in October 2012 and is a normal fuel shuffle RFO and not a complete full core offload outage. The spring 2014 RFO will be a planned full core offload outage; the plant is currently on an 18 month refueling cycle. The licensee stated that postponing the Code repair to the 2014 RFO would result in the clamp being installed for a total time period of 21 months (from July 11, 2012 to May 2014).

The licensee contended that changing the scope of the 2012 RFO to a full core offload to perform the ASME Code repair on the degraded weld is a significant hardship because the significant planning and scheduling resources are required to expedite a normal 18 month planning process. The licensee explained that in addition to the hardship of implementing a full core offload, the additional work and evaluation associated with schedule development and implementation is expected to be challenged by altering the outage schedule at this time. While these challenges can be managed, the measures required to perform an ASME Code repair during the 2012 RFO do not provide a compensating increase in the level of quality or safety to account for the impacts that would be recognized.

3.5 Proposed Alternative and Basis for Use

As an alternative to the ASME Code repair, the licensee installed a mechanical clamp on the degraded weld in accordance with the ASME Code, Section XI, Appendix IX to stop the leak and provide structural integrity to the degraded weld until the next (2012) RFO. However, because the 2012 RFO is not a full core offload, the licensee cannot perform the Code repair and requested to leave the clamp on the degraded weld for an additional fuel cycle. Therefore, Relief Request Number ISI-09 asks relief from Article IX-1000(a) of the ASME Code, Section XI, which states that mechanical clamping devices used as a piping pressure boundary may remain in service until the next RFO, at which time the defect shall be removed or reduced to an acceptable size.

3.6 Duration of Proposed Alternative

This relief request is applicable to R.E. Ginna Nuclear Power Plant's Fifth Interval ISI Program. The duration of this relief request is through the 2014 Refueling Outage, Cycle Number 38 (May 2014). The licensee stated that the proposed alternative provides an acceptable level of quality and safety as a result of using the approved ASME Code Section XI Mandatory Appendix IX, Mechanical Clamping Devices for Class 2 and 3 Piping Pressure Boundaries that provides structural integrity and could be used for a period of up to 24 months.

3.7 The NRC Staff Evaluation

The NRC staff evaluated (1) the mechanical clamp design, (2) defect characterization, (3) the potential degradation mechanisms, (4) the maximum flaw the weld can tolerate without a pipe rupture, (5) the flood analysis, (6) monitoring, and (7) hardship.

Mechanical Clamp Design

The licensee designed the mechanical clamp in accordance with site design requirements, operating conditions, material properties, and the ASME Code, Section XI, Mandatory Appendix IX. The licensee also injected sealant between the clamp and the pipe to stop the leak. The clamp and sealant material were selected for chemical compatibility with the potassium chromate in the piping system. The licensee analyzed the sealant material to ensure that it would not have adverse effects on the water chemistry of the system. The licensee stated that the sealant material is a silicon-elastomer material with fiber reinforcement, which has good chemical compatibility with potassium chromate. The clamp and the existing half-coupling are made of carbon steel to minimize the potential for galvanic corrosion. In addition, the thermal well side of the clamp has a stainless steel/carbon steel interface. The licensee noted that there is no source of fluid or electrolytes in that location to initiate galvanic corrosion.

The clamp is designed to cover the weld utilizing setscrews on the header side, and has mechanical crunch teeth to create a mechanical seal on both ends. In accordance with the ASME Code, Section XI, Appendix IX, the friction force that the set screws impart on the half-coupling exceeds the maximum longitudinal load by a safety factor of 5. The licensee analyzed the clamp with ejection forces of the worst-case failure and determined that the clamp would maintain pressure boundary structural integrity of the branch connection in the event of a worst-case circumferential failure.

The licensee stated that the clamp design ensures that any potential propagation of the flaw would be contained within the clamp. In the improbable event that the flaw grew beyond the value provided in the bounding calculation, the clamp has been designed and evaluated to accommodate the design forces of the CCW piping system. In addition to the required frictional forces to prevent ejection, the licensee evaluated the clamp shell and enclosure fasteners for all loading conditions including pressure, deadweight, thermal expansion, and earthquake.

The licensee also assessed the impacts of the additional weight of the clamp on the existing piping including seismic loading. To analyze the seismic effects, the licensee applied the entire weight of the clamp, including sealant and hardware, as a lump sum on the far edge of the clamp to bound the center of gravity effects. The licensee has demonstrated that the resulting stresses due to all applied loadings were below ASME Code, Section III, allowable stresses for all components and that there were no aggregate impacts on the 1-inch pipe, half coupling, the main 14-inch pipe, or the nearest lateral and vertical pipe supports.

Based on the above licensee's evaluation, the NRC staff concludes that the mechanical clamp will provide reasonable assurance of the structural integrity of the degraded weld and associated pipe until the 2014 RFO because the licensee has adequately demonstrated that the clamp is designed to satisfy the requirements of the ASME Code, Section III, and Section XI, Mandatory Appendix IX and that the clamp and the sealant material are chemically compatible with the system piping material.

Defect Characterization

Based on its VT-1 visual examination, the licensee stated that the defect occurred during the plant construction. The original plant construction code at Ginna Station is the American National Standards Institute (ANSI) B31.1, 1955 Edition. In accordance with this code, the fabrication inspection requirement for this line was a workmanship visual examination. Due to

the small pipe size, no radiographic (RT) examination was required to identify this original construction defect. By letter dated September 18, 2012, the licensee stated that it removed the clamp and performed a supplemental RT at the degraded weld on September 12, 2012. The licensee confirmed during the initial visual examination results that the defect was caused by a lack of fusion in the weld. The licensee noted that the defect length was 0.25 inches based on RT which did not identify any other flaws at the degraded weld. Besides RT, the licensee also attempted ultrasonic testing on the degraded weld. However, as a result of the pipe configuration, the licensee was not able to examine the area of concern. The licensee was able to determine that there is no indication on the pipe base metal on the instrument side. The NRC staff finds that the licensee has performed adequate defect characterization because licensee's supplemental RT confirmed that the detected flaw was caused by the lack of fusion.

Potential Degradation Mechanisms

The licensee performed failure modes and effects analysis (FMEA) of all potential degradation mechanisms that could have caused the lack of fusion defect to grow through the remaining thickness of the weld and identified the following probable failure mechanisms that could not immediately be eliminated based on available data: vibration induced fatigue crack of the existing defect, corrosion propagation of the existing defect, and external force propagation of the existing defect. The licensee could not conclusively determine the cause of the leak from the above three degradation mechanisms without performing a destructive examination of the degraded weld. The licensee reasoned that it is unlikely that the external force caused the existing defect to propagate because the flaw is located at the bottom of the pipe which would require upward force for a tensile failure. Therefore, the licensee included vibration and corrosion but not external force in its flaw tolerance evaluation. The NRC staff noted that the licensee' radiographic testing identified the presence of the defect that the licensee has attributed to lack of fusion at the time of construction and the FMEA is to identify the potential degradation mechanisms causing that defect to grow through the remaining weld thickness. The NRC staff finds that the licensee has performed a reasonable FMEA to exclude the probable causes of the defect and therefore it is likely that vibration and corrosion are the most probable degradation mechanisms.

Flaw Tolerance Evaluation

For the vibration degradation mechanism, the licensee calculated an applied alternating stress intensity of 6.7 ksi which is below the material endurance limit of 12.5 ksi for the carbon steel base metal. This shows that vibration induced stress levels are insufficient to allow crack initiation or propagation to occur. The licensee concluded that fatigue cracking cannot be initiated nor can it propagate due to vibration, given the small stress field present in this portion of the CCW system. The fracture mechanics evaluation also concluded that the maximum applied stress intensity factor of 2.9 ksi/in^2 does not exceed the ASME Code allowable stress intensity factor of 14.1 ksi/in^2 for carbon steel. In addition, the maximum stress intensity factor range (ΔK) of 3.77 ksi/in^2 is below the ΔK threshold for fatigue crack growth of 4.0 ksi/in^2. The licensee stated that one possible cause of the defect propagation is due to internal corrosion. In this scenario, the weld defect has grown to a pinhole leak by a form of corrosion, such as galvanic corrosion at the carbon-stainless fillet weld. The licensee noted that based on industry studies, typical galvanic corrosion rates of 10-30 mil/year have been noted at carbon steel/stainless steel dissimilar metal welds. The licensee used a bounding corrosion rate of 30 mil/year to calculate a metal loss of 0.053 inches over the 21 month period which was added to the crack growth calculation.

To address possible undetected subsurface cracks growing along the weld fusion line, the licensee assumed hypothetical circumferential flaws. The crack growth evaluations have shown that a 173-degree through-wall circumferential flaw and a 360-degree flaw with a depth per thickness ratio of 48 percent can be tolerated for the 21 months of operation without pipe rupture. Both of these analyses include metal loss due to the calculated bounding corrosion rate.

The NRC staff reviewed the licensee's applied alternating stress intensity factor for the vibration degradation mechanism and found it acceptable because the inputs and assumptions were appropriate and the results indicate the applied stress intensity factor is below the material endurance limit for the carbon steel base metal. In terms of the flaw evaluation, the NRC staff finds that the licensee has demonstrated by analysis that the degraded weld can tolerate a much larger flaw than the detected flaw size and that within the 21 months of operation, the detected flaw will not grow to affect the integrity of the weld. Therefore, the NRC staff finds that the licensee's flaw tolerance analysis is acceptable.

Flood Analysis

The licensee analyzed the impact of a circumferential pipe failure as part of the flood analysis. If the 1-inch line were to fail, the low-pressure and low-temperature water would spray onto the nearby pipes, heat exchanger, and possibly spray the "drip-proof" CCW pump motor casing. Based on the existing CCW Surge Tank level setpoints, a volume of 150 gallons could leak from the system prior to receiving a Main Control Room alarm. Section 9.2.2.4.1.4 of the Updated Final Safety Analysis Report states that the surge tank also maintains a positive suction head on the CCW pumps during normal and postaccident operation. Makeup water to the CCW system is normally supplied by the reactor makeup water system via a remotely operated valve in the auxiliary building. The makeup rate is sufficient to accommodate system leakage. The demineralized water system is also a makeup source utilizing manual valves. Installation of redundant water level sensors on the CCW surge tank ensures early warning and detection of leaks in the CCW system so that operator action can be taken to prevent damage to the reactor coolant pumps. Upon confirmation of a lowering Surge Tank level, the licensee would secure both CCW pumps and commence a plant shutdown in accordance with approved operating procedures. Upon securing the CCW pumps, the maximum volume of water which could leak out of this location is approximately 2000 gallons, which is significantly less than the values assumed in the Auxiliary Building flooding analysis. Therefore, the licensee stated that the maximum flooding from a highly improbable circumferential pipe failure is bounded by the existing internal flood analysis. The NRC staff finds that the licensee's flood analysis of the degraded weld does consider the worst case pipe failure and that the maximum leakage from the pipe failure is bounded by the design basis Auxiliary Building flood analysis. Therefore, the licensee has adequately addressed the potential for flooding caused by the degraded weld.

Monitoring

In accordance with the requirements of the ASME Code, Section XI, Appendix IX, Article 6000, the licensee is and will be performing a weekly inspection of the clamp for leakage. The licensee will also perform the quarterly volumetric examinations in accordance with Article IX-6000(a) unless precluded by the clamp configuration. The licensee stated that this monitoring will be in effect until the mechanical clamp is removed and a permanent repair is performed.

The NRC staff finds that the licensee will comply with the monitoring requirements of the ASME Code, Section XI, Appendix IX. Therefore, the proposed monitoring is acceptable.

Hardship

The licensee stated that changing the scope of the 2012 RFO to a full core offload to perform the ASME Code repair on the degraded weld is a significant hardship because the significant planning and scheduling resources are required to expedite a normal 18 month planning process. In addition, the additional work and evaluation associated with schedule development and implementation is expected to be challenged by altering the outage schedule at this time. The NRC staff believes that the above hardship or difficulty may not be particularly severe and that the licensee should be able to manage the changes. However, the NRC staff finds that requiring the licensee to repair the degraded weld during the 2012 RFO as opposed to the 2014 RFO does not provide a compensating increase in the level of quality or safety, accounting for the impacts that would be caused to the licensee.

4.0 CONCLUSION

As set forth above, the NRC staff concludes that the proposed alternative provides reasonable assurance of structural integrity and leak tightness of the clamped temperature element TE-621. The NRC staff concludes that complying with the specified ASME Code requirement would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety. Accordingly, the NRC staff concludes that the licensee has adequately addressed all of the regulatory requirements set forth in 10 CFR 50.55a(a)(3)(ii) and is in compliance with the requirements of the ASME Code, Section XI.

Therefore, the NRC staff authorizes the licensee's Relief Request Number ISI-09 at the R.E. Ginna Nuclear Power Plant until a permanent repair can be implemented during the 2014 RFO, Cycle Number 38, scheduled for May 2014.

All other ASME Code, Section XI requirements for which relief has not been specifically requested and approved in this relief request remain applicable, including third party review by the Authorized Nuclear Inservice Inspector.

Principal Contributors: J. Tsao
C. Sydner

Date: October 16, 2012

If you have any questions, please contact the R.E. Ginna Project Manager, Mohan Thadani, at (301) 415-1476.

Sincerely,

/RA/

George Wilson, Chief
Plant Licensing Branch I-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-244

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