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

March 14, 2016

Mr. Ken J. Peters
Senior Vice President and
Chief Nuclear Officer (Acting)
Attention: Regulatory Affairs
Luminant Generation Company LLC
P.O. Box 1002
Glen Rose, TX 76043

SUBJECT: COMANCHE PEAK NUCLEAR POWER PLANT, UNIT 1– RELIEF REQUEST 1B3-3, ALTERNATIVE TO THE ASME CODE, SECTION XI, EXAMINATION REQUIREMENTS FOR REACTOR PRESSURE VESSEL COLD-LEG WELD INSPECTION FREQUENCY (CAC NO. MF6125)

Dear Mr. Peters:

By letter dated April 20, 2015 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML15119A216), as supplemented by letters dated October 15, 2015, and February 15, 2016 (ADAMS Accession Nos. ML15300A013 and ML16057A309, respectively), Luminant Generation Company, LLC (the licensee) submitted Relief Request (RR) 1B3-3 to the U.S. Nuclear Regulatory Commission (NRC) for Comanche Peak Nuclear Power Plant (CPNPP), Unit 1, for the third 10-year inservice inspection program interval. Relief Request 1B3-3 requests relief from certain requirements of the American Society of Mechanical Engineers (ASME) *Boiler and Pressure Vessel Code* (Code) associated with the weld inspection frequency for the reactor pressure vessel cold leg at CPNPP, Unit 1.

Specifically, pursuant to Title 10 of the *Code of Federal Regulations* (10 CFR) 50.55a(z)(2), the licensee requested to use the proposed alternative for the reactor pressure vessel cold-leg weld inspection frequency as specified in Code Case N-770-1 for a period not to exceed 9 years, instead of a period of not to exceed 7 years, on the basis that complying with the specified requirement would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety.

The NRC staff has reviewed the subject request and concludes, as set forth in the enclosed safety evaluation, that for RR 1B3-3, the proposed alternative provides reasonable assurance of structural integrity of the affected components, and that complying with the specified 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(z)(2) for RR 1B3-3. Therefore, the NRC staff authorizes the use of alternative RR 1B3-3 for CPNPP, Unit 1, until startup from the spring 2019 refueling outage (1RF20).

All other ASME Code, Section XI, requirements for which relief was not 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 Ms. Margaret Watford of my staff at 301-415-1233 or via e-mail at Margaret.Watford@nrc.gov.

Sincerely,

Robert J. Pascarelli, Chief
Plant Licensing Branch IV-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-445

Enclosure:
Safety Evaluation

cc w/encl: Distribution via Listserv

UNITED STATES
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SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELIEF REQUEST 1B3-3

LUMINANT GENERATION COMPANY LLC

COMANCHE PEAK NUCLEAR POWER PLANT, UNIT 1

DOCKET NO. 50-445

1.0 INTRODUCTION

By letter dated April 20, 2015 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML15119A216), as supplemented by letters dated October 15, 2015, and February 15, 2016 (ADAMS Accession Nos. ML15300A013 and ML16057A309, respectively), Luminant Generation Company, LLC (the licensee) submitted Relief Request (RR) 1B3-3 to the U.S. Nuclear Regulatory Commission (NRC) for Comanche Peak Nuclear Power Plant (CPNPP), Unit 1, for the third 10-year inservice inspection (ISI) program interval. Relief Request 1B3-3 requests relief from certain requirements of the American Society of Mechanical Engineers (ASME) *Boiler and Pressure Vessel Code* (Code) associated with the weld inspection frequency for the reactor pressure vessel (RPV) cold leg at CPNPP, Unit 1.

Specifically, pursuant to Title 10 of the *Code of Federal Regulations* (10 CFR) 50.55a(z)(2), the licensee requested to use the proposed alternative for the RPV cold-leg weld inspection frequency as specified in Code Case N-770-1 for a period not to exceed 9 years, instead of a period of not to exceed 7 years, on the basis that complying with the specified requirement would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety.

2.0 REGULATORY EVALUATION

The ISI of ASME Code Class 1, 2, and 3 components is to be performed in accordance with Section XI, "Rules for Inservice Inspection of Nuclear Power Plant Components," of the ASME Code and applicable editions and addenda as required by 10 CFR 50.55a(g), except where specific written relief has been granted by the Commission.

Paragraph 10 CFR 50.55a(g)(6)(ii) states that the Commission may require the licensee to follow an augmented ISI program for systems and components for which the Commission deems that added assurance of structural reliability is necessary. Paragraph 10 CFR 50.55a(g)(6)(ii)(F) requires, in part, augmented inservice volumetric inspection of Class 1 piping and nozzle dissimilar welds (DM) butt welds of pressurized-water reactors in accordance with ASME Code Case N-770-1, subject to the conditions specified in paragraphs (2) through (10) of 10 CFR 50.55a(g)(6)(ii)(F).

Enclosure

Alternatives to requirements under 10 CFR 50.55a(g) may be authorized by the NRC pursuant to 10 CFR 50.55a(z)(1) or 10 CFR 50.55a(z)(2). In proposing alternatives or requests for relief, the licensee must demonstrate that: (1) the proposed alternatives would provide an acceptable level of quality and safety; or (2) 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 analysis of the regulatory requirements, and subject to the following technical evaluation, the NRC staff concludes that regulatory authority exists for the licensee to request the use of an alternative and the NRC to authorize the alternative proposed by the licensee on the basis that compliance with the specified requirements would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety. Accordingly, the NRC staff has reviewed and evaluated the licensee's request pursuant to 10 CFR 50.55a(z)(2).

3.0 TECHNICAL EVALUATION

3.1 Licensee Relief Request

3.1.1 Component Identification

The affected components are as follows:

Weld TBX-1-4100-14, Loop 1 cold-leg nozzle to safe-end weld
Weld TBX-1-4200-14, Loop 2 cold-leg nozzle to safe-end weld
Weld TBX-1-4300-14, Loop 3 cold-leg nozzle to safe-end weld
Weld TBX-1-4400-14, Loop 4 cold-leg nozzle to safe-end weld

3.1.2 Code Requirements for Which Relief is Requested

Volumetric inspection of RPV inlet cold-leg nozzle to safe-end DM welds of pressurized-water reactors are required in accordance with ASME Code Case N-770-1, subject to the conditions specified in paragraphs (2) through (10) of 10 CFR 50.55a(g)(6)(ii)(F). ASME Code Case N-770-1, Table 1, Inspection Item B requires volumetric examination of essentially 100 percent of each weld once every second inspection period not to exceed 7 years.

3.1.3 Licensee's Proposed Alternative and Duration of Relief

The licensee proposed a one-time extension to the Code Case N-770-1, Table 1, Inspection Item B, volumetric examinations from a period of "not to exceed 7 years" to a period of "not to exceed 9 years."

The licensee proposed an alternative to the regulatory requirement, which would reschedule the inspections from spring 2016 refueling outage to the spring 2019 refueling outage. This is a one-time extension inspection frequency request.

3.1.4 Licensee's Basis for Relief

The licensee stated that the relief request was due to the need to examine the RPV inlet cold-leg nozzle to safe-end welds from the inside surface of the weld. This requires access to the lower portion of the RPV to insert automated volumetric inspection equipment to perform the examination. As such, it would be necessary to remove the core barrel and other RPV internals. The core barrel is scheduled to be removed for inspection of the vessel shell welds and vessel internal inspection required by Electric Power Research Institute (EPRI) report 1022863, "Materials Reliability Program: Pressurized Water Reactor Internals Inspection and Evaluation Guidelines (MRP-227-A)," December 2011 (ADAMS Accession No. ML120170453), during the spring 2019 refueling outage. Requiring the additional removal of the core barrel and other internals during the spring 2016 refueling outage would result in additional radiological personnel dose and an additional heavy load lift in containment.

The licensee discussed in its submittal that the technical basis for the relief request included the considerations that: (1) there has been no service experience with cracking found in any RPV inlet cold-leg DM welds; (2) crack growth rates in RPV inlet cold-leg DM welds are slow; (3) the likelihood of initial cracking; crack growth and a subsequent through-wall leak is very small in these welds; and (4) the specific axial and circumferential flaw evaluation showing that any indication detected during the previous inspection, as well as any flaw size which could have been reasonably missed during RPV inlet cold-leg DM weld examination, would not grow to the allowable size flaw specified by ASME Code, Section XI, during the requested inspection interval. The licensee provided this technical basis to demonstrate that it is acceptable to extend the re-examination interval.

The licensee also stated that since primary water stress-corrosion cracking (PWSCC) is temperature dependent, it would be expected that hot-leg temperature welds would show evidence of crack initiation before cold-leg temperature welds, and no evidence of cracking has been identified in either hot-leg or cold-leg welds. Further, the cold-leg temperature welds that are the subject of this relief request were inspected in spring 2010 with volumetric techniques. No indications were identified in the welds.

The licensee provided a plant-specific flaw analysis for the RPV inlet cold-leg DM welds to support its proposal. The analysis was developed and based on the most recent guidance of EPRI report 1021023, "Materials Reliability Program: Primary Water Stress Corrosion Cracking (PWSCC) Flaw Evaluation Guidance (MRP-287)," December 2010 (publicly available at http://www.epri.com/abstracts/Pages/ProductAbstract.aspx?ProductId=000000000001021023). In development of the weld residual stresses, the licensee included the effects of a hypothetical 50 percent through-wall inside diameter surface weld repair. In summary, the licensee stated that the calculations show that in order for a flaw to have grown to a depth of 75 percent through-weld by the next inspection in spring 2019, an axial flaw would have had to have been 57 percent through-weld thickness or a circumferential flaw would have had to have been 33 percent through-weld thickness, during the previous inspection in spring of 2010. The licensee stated that based on the current inspection capabilities, the flaw sizes above are significantly larger than the theoretical flaw detection limit and the minimum size flaw actually detected during the previous inspection.

Finally, the licensee included a discussion on the probability of cracking or leakage from these welds. The licensee stated that there were analyses and sensitivity studies which showed the likelihood of cracking or through-wall leaks was very small and that more frequent inspections had only a small benefit in terms of risk. Further, the licensee noted that there was no operational experience of cracking in RPV inlet cold leg DM welds, and the number of indications in RPV hot-leg DM welds, a leading indicator of cracking due to temperature, was small when compared to the number of those welds in service.

Additionally, the licensee stated that volumetric inspection of the RPV inlet cold-leg nozzle to safe-end welds from the outside surface would be undesirable due to the welds being located inside a sandbox and covered with insulation. The sandbox was installed during original plant construction after all welding was completed.

Therefore, the licensee concluded that extending the required RPV inlet cold-leg DM weld volumetric examination until spring 2019 is justified.

As such, the licensee concluded that the technical basis was sufficient to ensure public health and safety by extending the inspection frequency of the RPV inlet cold-leg nozzle to safe-end DM welds from a maximum of 7 years to a new maximum of 9 years.

3.2 NRC Staff Evaluation

The NRC staff notes that the generic rules for the frequency of volumetric examination of DM butt welds were established to provide reasonable assurance of the leak tightness and structural integrity of the reactor coolant pressure boundary. As such, the staff finds that a plant-specific analysis could be used to provide a reasonable basis for inspection relief if the inspection frequency can be shown to maintain reasonable assurance of the leak tightness and structural integrity of the weld. As such, the staff reviewed the licensee's proposed alternative under the requirements of 10 CFR 50.55a(z)(2) that:

> Compliance with the specified requirements would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety.

The NRC staff reviewed the licensee's basis for hardship. Since the RPV inlet cold-leg DM welds are located in sandboxes, and inspection of the welds would require the licensee to remove the RPV core barrel only for these examinations in spring 2016 refueling outage (1RF18), the staff concludes that the licensee has a sufficient basis for hardship due to the expected radiological dose for the work.

Given the basis for hardship, the licensee's technical justification for the proposed alternative is that it provides reasonable assurance of structural integrity and leak tightness. The licensee also stated that no flaw of a size that could potentially have been missed during the 2010 refueling outage inspection could reasonably grow to an unacceptable size prior to the proposed inspection in 2019. The NRC staff reviewed the licensee's inspection results and flaw analysis to assess the acceptability of the proposed alternative.

The NRC staff reviewed each of the licensee's four bases: (1) there has been no service experience with cracking found in any RPV inlet cold-leg DM welds; (2) crack growth rates in

RPV inlet cold-leg DM welds are slow; (3) the likelihood of initial cracking, crack growth, and a subsequent through-wall leak is very small in these welds; and (4) the specific axial and circumferential flaw evaluation. The NRC staff notes that cracking has been identified in cold-leg temperature DM welds of smaller pipe size than the RPV inlet nozzle. Furthermore, cracking has been found in other locations at cold-leg temperatures in the reactor coolant pressure boundary. Additionally, reviews of the inspection data from those flaws have shown faster than average growth rates for the cold-leg temperatures. The staff notes that the RPV inlet cold-leg DM welds are made with weld materials susceptible to PWSCC and that this type of cracking can initiate and grow at cold-leg temperatures. As such, the staff concludes that for the licensee's items (1) – (3), there is insufficient basis to assume that cracking could not occur in these welds over time, and those flaws could not grow to a size that could challenge leak tightness or structural integrity.

However, the licensee's fourth basis, regarding the plant-specific axial and circumferential flaw evaluation, assumes that a hypothetical flaw could exist and provides an assessment of the potential growth of that flaw over time. The NRC staff reviewed the analysis and concluded that it could provide a basis to demonstrate leak tightness and structural integrity. Therefore, the staff focused its review on this aspect of the licensee's basis for the proposed alternative.

The licensee's flaw analysis is composed of a stress analysis and a flaw growth calculation. The NRC staff reviewed the licensee's stress analysis and found that it followed the recommendations of MRP-287 on effective weld residual stress calculations to address PWSCC flaw analysis. To add significant conservatism, a 50 percent inside surface weld repair 360 degrees around the circumference was simulated in the weld residual stress analysis. The fabrication sequence was simulated based on information provided in the plant-specific drawings. The staff also concluded that that the use of two stress paths, calculated for both hoop and axial stresses, was effective and consistent with NRC staff expectations. The staff reviewed the final proprietary stress analysis through the thickness of the weld and found both the hoop and axial residual stress curve contours were consistent with the MRP-287 analyses using similar geometries and fabrication methods. Based on its review, the NRC staff concluded that the licensee's plant-specific stress analysis for the subject welds to have conservative inputs and assumptions and, therefore, was adequate to be used in the flaw evaluation.

The NRC staff reviewed the licensee's previous inspection methods and results to assess the licensee's basis for assuming a maximum hypothetical initial flaw size during the 2010 outage. The basis included an ASME Code, Section XI, Appendix VIII demonstrated volumetric examination obtaining essentially 100 percent coverage that found no indications of surface connected flaws. The staff concluded that the licensee's qualified inspection techniques provide a reasonable basis that any flaw connected to the wetted surface with a size of 10 percent in depth or greater should have been identified. Also, the staff concluded that the licensee's data and supporting inspection results provided a reasonable basis for the initial flaw size assumptions.

The NRC staff assessed the licensee's proposed alternative by performing a series of flaw evaluations. The staff's flaw evaluations demonstrated that there is sufficient margin between the hypothetical maximum size of the postulated flaw after 9 years of growth and the ASME Code allowable flaw size, which supports the licensee's proposed alternative.

Therefore, based on the hardship of the increased radiological dose required to perform the required examinations due to the location of the RPV inlet cold-leg nozzle to safe-end DM welds in the sandboxes, and the licensee's flaw analysis demonstrating a sufficient safety margin, the NRC staff concludes that the licensee has provided an adequate technical basis to demonstrate that compliance with the requirements of 10 CFR 50.55a(g)(6)(ii)(F) for the volumetric inspection of the RPV inlet cold-leg nozzle to safe-end DM welds during the spring 2016 refueling outage would result in a hardship or unusual difficulty without a compensating increase in the level of quality and safety.

4.0 CONCLUSION

Based on the above, the NRC staff concludes that that the proposed alternative provides reasonable assurance of structural integrity of the subject components and that complying with the 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(z)(2). Therefore, the NRC staff authorizes the use of the proposed alternative RR 1B3-3 at CPNPP, Unit 1, until startup from the spring 2019 refueling outage (1RF20).

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

Principal Contributor: M. Audrain, NRR/DE/EPNB

Date: March 14, 2016

All other ASME Code, Section XI, requirements for which relief was not 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 Ms. Margaret Watford of my staff at 301-415-1233 or via e-mail at Margaret.Watford@nrc.gov.

Sincerely,

/RA/

Robert J. Pascarelli, Chief
Plant Licensing Branch IV-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-445

Enclosure:
Safety Evaluation

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