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

October 30, 2015

Mr. Rafael Flores
Senior Vice President and
Chief Nuclear Officer
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 1B3-4 FOR APPLICATION OF AN ALTERNATIVE TO THE ASME BOILER AND PRESSURE VESSEL CODE EXAMINATION REQUIREMENTS FOR REACTOR PRESSURE VESSEL HEAD PENETRATION WELD INSPECTION FREQUENCY FOR THE THIRD 10-YEAR INSERVICE INSPECTION INTERVAL (CAC NO. MF6132)

Dear Mr. Flores:

By letter dated April 22, 2015 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML15120A038), Luminant Generation Company LLC (the licensee) submitted Relief Request (RR) 1B3-4 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) interval.

Specifically, pursuant to Title 10 of the *Code of Federal Regulations* (10 CFR) 50.55a(z)(1), the licensee requested to use the proposed alternative to the examination frequency of American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code), Section XI, Code Case N-729-1, "Alternative Examination Requirements for PWR [Pressurized-Water Reactor] Reactor Vessel Upper Heads With Nozzles Having Pressure Retaining Partial-Penetration Welds, Section XI, Division 1," on the basis that the alternative examination provides an acceptable level of quality and safety.

The NRC staff has completed its review of the proposed alternative and based on the enclosed safety evaluation, the NRC staff concludes that the alternative mrthod proposed by the licensee in RR 1B3-4 will provide an acceptable level of quality and safety for the examination frequency requirements of the reactor vessel closure head. The NRC staff concludes that the licensee has adequately addressed all of the regulatory requirements set forth in 10 CFR 50.55a(z)(1). Therefore, the NRC staff authorizes the one-time use of RR 1B3-4 at CPNPP, Unit 1, for the duration up to, and including refueling outage 1RF22, which is scheduled to commence in the spring of 2022 and occur in the fourth 10-year ISI interval. The third 10-year ISI interval began on August 13, 2010, and ends on August 12, 2020. The fourth ISI interval begins on August 13, 2020, and ends on August 12, 2030.

All other requirements of the ASME Code, Section XI, and 10 CFR 50.55a(g)(6)(ii)(D) for which relief was not specifically requested and approved in the subject request remain applicable, including third-party review by the Authorized Nuclear Inservice Inspector.

If you have any questions, please contact Balwant K. Singal at 301-415-3016 or via e-mail at Balwant.Singal@nrc.gov.

Sincerely,

Michael T. Markley

Michael T. Markley, 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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UNITED STATES
NUCLEAR REGULATORY COMMISSION
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SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELIEF REQUEST 1B3-4

THIRD 10-YEAR INSERVICE INSPECTION INTERVAL

LUMINANT GENERATION COMPANY LLC

COMANCHE PEAK NUCLEAR POWER PLANT, UNIT 1

DOCKET NO. 50-445

1.0 INTRODUCTION

By letter dated April 22, 2015 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML15120A038), Luminant Generation Company LLC (the licensee) submitted Relief Request (RR) 1B3-4 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) interval.

Specifically, pursuant to Title 10 of the *Code of Federal Regulations* (10 CFR) paragraph 50.55a(z)(1), the licensee requested to use the proposed alternative to the examination frequency of American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code), Section XI, Code Case N-729-1, "Alternative Examination Requirements for PWR [Pressurized Water Reactor] Reactor Vessel Upper Heads with Nozzles Having Pressure Retaining Partial-Penetration welds, Section XI, Division 1," on the basis that the alternative examination provides an acceptable 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 ASME Code, Section XI, "Rules for Inservice Inspection of Nuclear Power Plant Components," and applicable editions and addenda as required by 10 CFR 50.55a(g), "Inservice inspection requirements," except where specific written relief has been granted by the Commission.

The regulations in 10 CFR 50.55a(g)(6)(ii), state, in part, that, "[t]he Commission may require the licensee to follow an augmented inservice inspection program for systems and components for which the Commission deems that added assurance of structural reliability is necessary."

The regulations in 10 CFR 50.55a(g)(6)(ii)(D), require, in part, that, "[a]ll licensees of pressurized water reactors must augment their inservice inspection program with ASME Code Case N-729-1, subject to conditions specified in paragraphs (g)(6)(ii)(D)(2) through (6)...."

Enclosure

Pursuant to 10 CFR 50.55a(a)(z), proposed alternatives to the requirements of 10 CFR 50.55a(g) may be used, when authorized by the Director, Nuclear Reactor Regulation, if the licensee demonstrates (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 above, the NRC staff determines that regulatory authority exists for the licensee to request and the Commission to authorize the proposed alternative requested by the licensee.

3.0 TECHNICAL EVALUATION

3.1 ASME Code Components Affected

The affected components are ASME Class 1, Reactor Vessel Upper Head (Closure Head) (RVCH) penetration nozzles and partial penetration welds, which are fabricated from Inconel SB-167 (Alloy 690) UNS N06690. The original CPNPP, Unit 1, RVCH which contained penetration nozzles and were manufactured with Alloys 600/82/182 materials, was replaced with a new RVCH using Alloys 690/52/152 material for the penetration nozzles during the refueling outage that occurred during the spring 2007.

The NRC staff acknowledges the nozzle J-groove welds are fabricated from ERNiCrFe-7 (UNS N06052) and ENiCrFe-7 (UNS W86152), 52/152 weld materials.

3.2 Duration of the Proposed Alternative

Since utilization of the proposed alternative will require the examination to be performed in fourth ISI interval, the proposed duration of the alternative will occur in the third and fourth 10-year ISI interval. The third ISI interval began August 13, 2010, and ends August 12, 2020. The fourth ISI interval begins August 13, 2020, and ends August 12, 2030.

3.3 ASME Code of Record

The ASME Code, Section XI, Code of record for the third 10-year ISI interval is the 2007 Edition with 2008 Addenda. The ASME Code, Section XI Code of record for the fourth 10-year ISI interval will be the one approved for use on August 13, 2020.

3.4 ASME Code and/or Regulatory Requirements

The regulations in 10 CFR 50.55a(g)(6)(ii)(D) require, in part, that licensees shall augment their ISI program in accordance with ASME Code Case N-729-1, subject to the conditions specified in paragraphs (2) through (6) of 10 CFR 50.55a(g)(6)(ii)(D). ASME Code Case N-729-1, Table 1, Inspection Item B4.40 requires volumetric/surface examination be performed within one inspection interval (nominally 10 calendar years) of its inservice date for a replaced RVCH. The required volumetric/surface examinations would thus have to be completed by spring 2016 (refueling outage 1RF18) in order to fulfill the requirements of ASME Code Case N-729-1.

3.5 Proposed Alternative

The licensee proposes to delay the next required volumetric/surface examination of the replacement RVCH for a period of approximately 5 years from its current inspection date. The licensee proposes to accomplish the inspection in accordance with ASME Code Case N-729-1 and 10 CFR 50.55a(g)(6)(ii)(D) during refueling outage 1RF22, which is scheduled for spring 2022. The NRC staff notes that the current required inspection date occurs in the plant's third ISI interval and the proposed inspection will be accomplished during the plant's fourth ISI interval.

3.6 Licensee's Basis for Use of the Proposed Alternative

The licensee's basis for use of the proposed alternative is comprised of the following: 1) the inspection interval in ASME Code Case N-729-1 is based on primary water stress-corrosion cracking (PWSCC) crack growth rates for Alloy 600/82/182, which are conservative compared to the lower crack growth rates for Alloy 690/52/152; 2) bare-metal visual examination conducted on the licensee's replacement RVCH in 2011; and 3) a plant-specific factor of improvement (FOI) analysis conducted by the licensee.

In addressing its first basis for use of the proposed alternative, the licensee stated that the inspection intervals contained in ASME Code Case N-729-1 for Alloy 600/82/182 are based on re-inspection years (RIY) equal to 2.25. This RIY was developed based on PWSCC crack growth rates as defined in the 75th percentile curve contained in Electric Power Research Institute (EPRI) Materials Reliability Program (MRP)-55, "Crack Growth Rates for Evaluating Primary Water Stress Corrosion Cracking (PWSCC) of Thick-Wall Alloy 600 Materials," November 2002, and MRP-115, "Crack Growth Rates for Evaluating Primary Water Stress Corrosion Cracking (PWSCC) of Alloy 82, 182, and 132 Welds," November 2004 (MRP-55 and MRP-115 are available to the public at www.epri.com). The PWSCC crack growth rates of Alloy 690/52/152 are significantly lower than those of Alloy 600/82/182 and is the basis for a limited extension of volumetric/surface examination frequency. The licensee's justification is based on: a) the lack of cracking in other 690 components such as steam generators and pressurizers in the approximately 20 years that alloy 690 has been in service in these components; b) the failure to observe cracking in inspections already performed in replacement RVCHs (13 of 40 replacement RVCHs have been examined which includes RVCHs which operate at higher temperatures than the RVCH under consideration); c) the similarity of the inspected RVCHs to the RVCH under consideration regarding configuration, manufacturing, design and operating conditions; and d) laboratory test data for alloys 690/52/152 as contained in EPRI MRP-375, "Technical Basis for Reexamination Interval Extension for Alloy 690 PWR Reactor Vessel Top Head Penetration Nozzles," February 2014 (available to the public at www.epri.com).

The results of the bare-metal visual examination performed in fall 2011 (refueling outage 1RF15) on the CPNPP, Unit 1, replacement RVCH in accordance with ASME Code Case N-729-1, Table 1, item B4.30 was used to address the second basis for use of the proposed alternative. This visual examination was performed by VT-2 qualified examiners on the outer surface of the RVCH including the annulus area of the penetration nozzles. This examination did not reveal any indications of nozzle leakage (e.g.,boric acid deposits) on the surface or near a nozzle penetration. The licensee also indicated that this examination will be performed again in the upcoming refueling outage 1RF18 scheduled to commence in the spring of 2016. Also, no alternative examination processes are proposed to those required by ASME Code Case N-729-1, as conditioned by 10 CFR 50.55a(g)(6)(ii)(D). The visual (VT-2) examinations and acceptance criteria as required by item B4.30 of Table 1 of ASME Code Case N-729-1 are not affected by this request and will continue to be performed on a frequency not to exceed every 5 calendar years.

The results of the plant-specific calculation of the required FOI in the crack growth rate of Alloy 690/52/152 as compared to the crack growth rate of Alloy 600/82/182 was used to address the third basis for use of the proposed alternative. As inputs to the calculation, the licensee used the actual operating temperature of the RVCH for CPNPP, Unit 1, and conservatively assumed that calendar years were equal to effective full-power years (EFPYs). Based on this calculation, the licensee determined that a FOI of 2.5 was required to meet the proposed and desired inspection interval of 15 calendar years. Since the required FOI (2.5) was smaller than the FOI of 20 which bounded most of the MRP-375 data for alloy 690/52/152, the use of a FOI of 2.5 would not result in a reduction in safety and was, therefore, justified.

Based on the above, it was concluded that the licensee's analysis showed significant margin to ensure that Alloy 690 nozzle base and Alloy 52/152 weld materials used in the CPNPP, Unit 1, replacement RVCH provide for a reactor coolant system pressure boundary where, by analysis and by years of positive industry experience, the potential for PWSCC has been shown to be remote. Hence, the licensee determined the technical basis to be sufficient to extend the inspection frequency of the RVCH nozzle at CPNPP, Unit 1, from a maximum of 10 years to a new maximum of 15 years.

3.7 NRC Staff Evaluation

In evaluating the technical sufficiency of the licensee's proposed alternative (i.e., a one-time extension of the volumetric/surface examination interval contained in ASME Code Case N-729-1 from 10 years to not longer than 15 years), the NRC staff considered each of the three aspects of the licensee's basis for use of the proposed alternative.

Due to concerns about PWSCC, many PWR plants in the United States and overseas have replaced RVCHs containing Alloy 600/182/82 nozzles with RVCHs containing Alloy 690/152/52 nozzles. The inspection frequencies developed in ASME Code Case N-729-1 for RVCH penetration nozzles using Alloy 600/182/82 were developed based, in part, on materials crack growth rate equations documented in EPRI MRP-55 and EPRI MRP-115. The licensee's application provided information and data regarding the more PWSCC-resistant materials, Alloy 690/152/52, and calculations to demonstrate an FOI of these materials versus the Alloy 600/82/182 materials. This FOI would then provide the basis for the extension of the ISI frequency requested by the licensee in its proposed alternative.

In evaluating the licensee's first technical basis for use of the proposed alternative, the NRC staff notes that the licensee based its evaluation on the data provided by EPRI MRP-375. This document, in part, summarizes Alloy 690/152/52 crack growth rate data from various sources to develop FOIs for the crack growth rate equations provided in EPRI MRP-55 and EPRI MRP-115. While the NRC staff determines the licensee's justifications and/or interpretations are reasonable, EPRI MRP-375 is not an NRC-approved document. Therefore, since the licensee did not request review and approval of EPRI MRP-375 for this proposed alternative, the NRC staff did not use the data from this document to review the licensee's relief request. A detailed review of the data provided in EPRI MRP-375 will be performed by an international group of experts as part of an Alloy 690 Expert Panel, which is currently scheduled to complete its review in the 2016-2017 timeframe.

In the interim, the NRC staff review will rely upon Alloy 690/152/52 crack growth rate data from two NRC contractors: Pacific Northwest National Laboratory (PNNL) and Argonne National Laboratory (ANL). This data from these two contractors is documented in a data summary report and can be found under ADAMS Accession No. ML14322A587. This confirmatory research regarding Alloy 690/52/152 crack growth rates, performed by PNNL and ANL, generally supports the information provided by the licensee in its application dated April 22, 2015, that the crack growth rate of Alloy 690/52/152 is more crack-resistant but differs from the EPRI MRP-375 crack growth rate data in some respects.

The PNNL and ANL data summary report includes crack growth rate data up to approximately 20 percent cold work, based on the observation of local strains in welds and weld dilution zone data. However, the NRC staff did not consider the weld dilution zone data in its assessment because the limited weld dilution zone data that is currently available has shown higher crack growth rates than are commonly observed for Alloy 690/152/52 materials. The high crack growth rates in weld dilution zones may be due to the reduced chromium present in these areas. The NRC staff excluded the weld dilution zone data from this analysis due to the limited number of data points available, the variability in results, and due to the limited area of continuous weld dilution through which flaws grow. For example, in the case of the highest measured crack growth rates, a flaw would have to travel in the heat affected zone of a J-groove weld along the low alloy steel head interface. It is not fully apparent to the NRC staff how accelerated crack growth in very small areas of weld dilution zone would result in a significantly increased probability of leakage or component failure during a relatively short extension of the required inspection interval. Exclusion of this data may be reevaluated as additional data become available, a better understanding of the existing data is obtained, or if a longer extension of the inspection interval is requested. Therefore, the NRC staff concludes that the impact of these weld dilution zone crack growth rates on the change in volumetric inspection frequency, as requested by the licensee's proposed alternative, is not considered to be relevant for this specific relief request.

In evaluating the licensee's second basis for use of the proposed alternative, the NRC staff concludes that the past bare-metal visual examination on the head under consideration is a reasonable means to demonstrate the absence of leakage through the nozzle/J-groove weld prior to the time the examination was conducted. The NRC staff also concludes that performance of future bare-metal visual examinations in accordance with the ASME Code Case N-729-1 is adequate to demonstrate the absence of leakage at or prior to the time the examinations are conducted. Finally, the NRC staff concludes that the proposed alternative's frequency for bare-metal visual examinations, in conjunction with the new frequency of volumetric examinations, is sufficient to provide reasonable assurance of the structural integrity of the RVCH.

In evaluating the licensee's third basis for use of the proposed alternative, the NRC concludes that based on the NRC staff calculation, the licensee's calculated FOI of 2.5, to support an extension of the ASME Code Case N-729-1 inspection frequency of 2.25 RIY to 15 calendar years, is acceptable. The NRC staff also concludes that the application of an FOI of 2.5 to the 75th percentile curves in EPRI MRP-55 and EPRI MRP-115 bounded essentially all of the NRC data included in the PNNL and ANL data summary report. Therefore, the NRC staff concludes that this analysis supports the licensee's justification that a volumetric inspection interval for the RVCH for CPNPP, Unit 1, of not more than 15 calendar years does not pose a higher risk than that associated with an Alloy 600/182/82 RVCH inspected at intervals of 2.25 RIY. Hence, the NRC staff concludes that the licensee's technical basis for the proposed alternative provide an acceptable level of quality and safety as required by 10 CFR 50.55a(z)(1) and is acceptable.

4.0 CONCLUSION

As set forth above, the NRC staff has determined that the alternative method proposed by the licensee in RR 1B3-4 will provide an acceptable level of quality and safety for the examination frequency requirements of the RVCH. Accordingly, the NRC staff concludes that the licensee has adequately addressed all of the regulatory requirements set forth in 10 CFR 50.55a(z)(1) for the proposed alternative. Therefore, the NRC staff authorizes the one-time use of RR 1B3-4 at CPNPP, Unit 1, for the duration up to and including the 1RF22 refueling outage, which is scheduled to commence in the spring of 2022 and occur in the fourth 10-year ISI interval.

All other requirements of the ASME Code, Section XI, and 10 CFR 50.55a(g)(6)(ii)(D) for which relief was not specifically requested and approved in the subject request remain applicable, including third-party review by the Authorized Nuclear Inservice Inspector.

Principal Contributor: Donald Becker, NRR/DE/EPNB

Date: October 30, 2015

All other requirements of the ASME Code, Section XI, and 10 CFR 50.55a(g)(6)(ii)(D) for which relief was not specifically requested and approved in the subject request remain applicable, including third-party review by the Authorized Nuclear Inservice Inspector.

If you have any questions, please contact Balwant K. Singal at 301-415-3016 or via e-mail at Balwant.Singal@nrc.gov.

Sincerely,

/RA/

Michael T. Markley, 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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ADAMS Accession No. ML15259A004 *** Memo dated August 12, 2015**

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