NRR-PMDAPEm Resource

From: Singal, Balwant
Sent: Tuesday, November 08, 2016 8:58 AM
To: Muilenburg William T (wimuile@WCNOC.com)
Cc: Hafenstine Cynthia R; Alley, David; Pascarelli, Robert; Tsao, John
Subject: Wolf Creek Generating Station - Verbal Authorization for Relief Requests I4-03 and IR-04 (CAC No. MF8456)
Attachments: Wolf Creek verbal auth I4R-03 11-03-2016 Rev 2.docx; Wolf Creek verbal auth I4R-04 10-17-2016.docx

By telephone conference call on November 7, 2016, the U.S. Nuclear Regulatory Commission (NRC) staff provided verbal authorization to Wolf Creek Nuclear Operating Corporation (WCNOC, the licensee) for the subject relief requests based on the attached explanation.

Participants:

NRC

David Alley, Branch Chief (provided technical justification)
Robert Pascarelli, Branch Chief (provided authorization)
John Tsao, Senior Materials Engineer
Balwant Singal, Senior Project Manager

WCNOC

Jaime H. McCoy, Vice-President, Engineering
Dennis E. Tougaw, Engineer V, Engineering Programs
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Reece D. Hobby III, Licensing Engineer V, Licensing
Cynthia R. Hafenstine, Manager Regulatory Affairs

Please let us know if you have any questions.

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VERBAL AUTHORIZATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION
RELIEF REQUEST I4R-03
ALTERNATIVE TO USE VOLUMETRIC LEAK PATH FOR SUPPLEMENTAL EXAMS
WOLF CREEK GENERATING STATION
WOLF CREEK NUCLEAR OPERATING CORPORATION
DOCKET NUMBER 50-482

Technical Evaluation read by David Alley, Chief of the Component Performance, Non-Destructive Examination, and Testing Branch, Office of Nuclear Reactor Regulation

By letter dated October 11, 2016, as supplemented by letters dated October 14, October 20, and November 1, 2016, Wolf Creek Nuclear Operating Corporation (the licensee) submitted Relief Request I4R-03 for the alternate examination of all 78 control rod drive mechanism (CRDM) nozzle penetration welds at the Wolf Creek Generating Station.

The licensee proposed (a) to perform a volumetric leak path assessment of each penetration nozzle in lieu of the surface leak path assessment required by Paragraph -3200(b) of ASME Code Case N-729-1, and (b) if an unacceptable indication by the leak path assessment or volumetric exam is identified, the licensee will revert to the requirements of Code Case N-729-1 and 10 CFR 50.55a(g)(6)(ii)(D). The licensee made this request in accordance with the requirements of 10 CFR 50.55a(z)(2), such 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 finds that while the demonstrated volumetric leak path is not equivalent to a fully qualified surface leak path assessment, the licensee identified sufficient operational experience, technical basis and radiological dose hardship to show that regulatory compliance would result in hardship without a compensating increase in the level of quality and safety.

For operating experience, the licensee showed that there has been no previous identified cracking or leakage identified from the CRDM nozzle penetrations or welds of the upper head at Wolf Creek. The NRC staff noted that while this fact does not preclude the possibility of cracking to be found as the plant continues to age, plants which have previously identified cracking are more likely to see subsequent and more significant cracking in the future. Given the lack of the initial cracking being identified in the nozzle heats of material, at the operating temperatures of Wolf Creek, the NRC found that the potential for significant cracking this outage was less likely.

For technical basis, the licensee identified that their inspection would be in compliance with the Wesdye Technical Justification Document showing an effective demonstration of the volumetric leak path technique. The NRC has accepted the use of a demonstrated volumetric leak path as part of the upper head inspection program under 10 CFR 50.55a(g)(6)(ii)(D). The licensee also referenced NUREG/CR-7142, Ultrasonic Phased Array Assessment of the Interference Fit and Leak Path of the North Anna Unit 2 Control Rod Drive Mechanism Nozzle 63 with Destructive Validation, which found, in part, the use of a properly focused 0 degree probe could detect a leakage path under low leakage rates during operation that led to minimal wastage of the upper head low alloy steel. While the NRC staff did not find that the volumetric leak path assessment was equivalent to a qualified surface leak path assessment, the information does

demonstrate the effectiveness of the volumetric leak path examination to detect low leakage rates, as performed in accordance with the licensee's proposed alternative.

For hardship, the licensee noted that a qualified surface leak path assessment could be performed in two manners that would require both additional radiological dose and time versus the performance of a volumetric leak path assessment. The NRC staff found both of these conditions to be of sufficient hardship given the operational experience and technical adequacy of the licensee's proposed alternative versus the regulatory requirement.

Therefore, the NRC staff finds that the licensee's proposed alternative provides reasonable assurance of structural integrity until the next scheduled examination, and that compliance with the surface examination requirements of Paragraph -3200(b) of ASME Code Case N-729-1, for the subject welds, would result in hardship without a compensating increase in the level of quality and safety.

Authorization read by Robert Pascarelli, Chief of the Plant Licensing Branch IV-1, Office of Nuclear Reactor Regulation

As Chief of the Plant Licensing Branch IV-1, Office of Nuclear Reactor Regulation, I concur with the Component Performance, Non-Destructive Examination, and Testing Branch's determinations.

The NRC staff concludes that the proposed alternative provides reasonable assurance of structural integrity of all 78 CRDM penetration nozzles such that complying with the 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(z)(2) and 10 CFR 50.55a(g)(6)(ii)(D). Therefore, the NRC staff authorizes the use of relief request I4R-03 at the Wolf Creek Generating Station during the current refueling outage subject to the licensee's proposed alternative that if an unacceptable indication by the leak path assessment or volumetric exam is identified, the licensee will revert to the requirements of Code Case N-729-1 and 10 CFR 50.55a(g)(6)(ii)(D).

All other requirements of ASME Code, Section XI, for which relief was not specifically requested and authorized by the NRC staff remain applicable, including the third party review by the Authorized Nuclear In-service Inspector.

This verbal authorization does not preclude the NRC staff from asking additional clarification questions regarding Relief Request I4R-03, while preparing the subsequent written safety evaluation.

VERBAL AUTHORIZATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION
RELIEF REQUEST I4R-04
ALTERNATE EXAMINATION OF CONTROL ROD DRIVE MECHANISM
NOZZLE PENETRATIONS
WOLF CREEK GENERATING STATION
WOLF CREEK NUCLEAR OPERATING CORPORATION
DOCKET NUMBER 50-482

Technical Evaluation read by David Alley, Chief of the Component Performance, Non-Destructive Examination, and Testing Branch, Office of Nuclear Reactor Regulation

By letter dated October 11, 2016, as supplemented by letters dated October 14, October 20, and November 1, 2016, Wolf Creek Nuclear Operating Corporation (the licensee) submitted Relief Request I4R-04 for the alternate examination of control rod drive mechanism (CRDM) nozzle penetration numbers 77 and 78 at the Wolf Creek Generating Station.

The licensee proposed (a) an alternate examination distance for CRDM nozzle numbers 77 and 78 in lieu of the required examination distance per ASME Code Case N-729-1 as conditioned by 10 CFR 50.55a(g)(6)(ii)(D) and (b) not to perform the surface examination of the portion of the CRDM nozzle below the J-groove weld as required by 10 CFR 50.55a(g)(6)(ii)(D)(3). The licensee made this request in accordance with the requirements of 10 CFR 50.55a(z)(2), such 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 finds that the proposed examination distances are acceptable for CRDM nozzle numbers 77 and 78. This is based on the validity of the licensee's stress analysis and fracture mechanics calculation, demonstrating that within four refueling cycles, a potential flaw that initiates in the unexamined zone (below the J-groove weld) of the CRDM nozzle numbers 77 and 78 will not propagate into the J-groove weld. At the end of every fourth refueling cycle, the licensee will perform an examination to confirm the structural integrity of CRDM nozzles 77 and 78.

The NRC staff finds the licensee's hardship justification is acceptable because of the considerable radiation dose and the nozzle configuration that are not conducive for the required examination.

The NRC staff finds that the licensee's proposed alternative examination distances for CRDM penetration nozzle numbers 77 and 78 provides reasonable assurance of structural integrity and leak tightness until the next scheduled examination, and that compliance with the surface examination requirements of 10 CFR 50.55a(g)(6)(ii)(D)(3) would result in hardship without a compensating increase in the level of quality and safety.

Authorization read by Robert Pascarelli, Chief of the Plant Licensing Branch IV-1, Office of Nuclear Reactor Regulation

As Chief of the Plant Licensing Branch IV-1, Office of Nuclear Reactor Regulation, I concur with the Component Performance, Non-Destructive Examination, and Testing Branch's determinations.

The NRC staff concludes that the proposed alternative provides reasonable assurance of structural integrity of the CRDM penetration nozzles numbers 77 and 78 and that complying with the 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(z)(2) and 10 CFR 50.55a(g)(6)(ii)(D). Therefore, the NRC staff authorizes the use of relief request I4R-04 at the Wolf Creek Generating Station for the remainder of the fourth 10-year ISI interval, which ends on September 2, 2025.

All other requirements of ASME Code, Section XI, for which relief was not specifically requested and authorized by the NRC staff remain applicable, including the third party review by the Authorized Nuclear In-service Inspector.

This verbal authorization does not preclude the NRC staff from asking additional clarification questions regarding Relief Request I4R-04, while preparing the subsequent written safety evaluation.