WOLF CREEK
NUCLEAR OPERATING CORPORATION

Cleveland Reasoner
Vice President Engineering

August 21, 2014
ET 14-0026

U. S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, DC 20555

Reference: 1) Letter ET 14-0019, dated June 26, 2014, from C. O. Reasoner, WCNOC, to USNRC

2) Letter dated July 25, 2014, from C. F. Lyon, USNRC, to A.C. Heflin, WCNOC

Subject: Docket No. 50-482: Response to Request for Additional Information Regarding Request for Alternative I3R-11

Gentlemen:

Reference 1 provided Wolf Creek Nuclear Operating Corporation's (WCNOC) 10 CFR 50.55a Request Number I3R-11 for the Third Ten-Year Interval of WCNOC's Inservice Inspection (ISI) Program. Request I3R-11 requests relief from the pressure test requirements of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code, Section XI, IWC-5220.

Reference 2 provided a Nuclear Regulatory Commission (NRC) request for additional information (RAI) regarding 10 CFR 50.55a Request Number I3R-11. The attachment provides WCNOC's response to the questions in the RAI.

This letter contains no commitments. If you have any questions concerning this matter, please contact me at (620) 364-4171, or Mr. Steven R. Koenig at (620) 364-4041.

Sincerely,

Cleveland Reasoner

Cleveland Reasoner

COR/rlt

Attachment

cc: M. L. Dapas (NRC), w/a
C. F. Lyon (NRC), w/a
N. F. O'Keefe (NRC), w/a
Senior Resident Inspector (NRC), w/a

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NRR

Response to Request for Additional Information Regarding Request for Alternative I3R-11 to ASME Code Requirements

Reference 1 provided Wolf Creek Nuclear Operating Corporation's (WCNOC) 10 CFR 50.55a Request Number I3R-11 for the Third Ten-Year Interval of WCNOC's Inservice Inspection (ISI) Program. Request I3R-11 requests relief from the pressure test requirements of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel (BPV) Code, Section XI, IWC-5220. Reference 2 provided a Nuclear Regulatory Commission (NRC) request for additional information (RAI) regarding 10 CFR 50.55a Request Number I3R-11. The specific NRC questions are provided below in italics followed by the WCNOC response.

1. *Please discuss whether or not the subject piping can be pressure tested in accordance with the ASME Code, Section XI, IWA-5000 and IWC-5000 before removing the RPV head in the beginning of a refueling outage. If yes, the proposed relief request would not be needed. If no, please discuss the hardship and potential personnel radiation exposure, including an estimate for person-roentgen equivalent man (rem) exposure with consideration of as low as reasonably achievable (ALARA) for the system leakage test performed in accordance with IWC-5221 before removing the RPV head in the beginning of a refueling outage.*

Response: The following response will first describe alternative test options beyond those already discussed in the WCNOC Request I3R-11, which could possibly be used to pressure test the leak-off piping to a higher pressure although still not at the reactor coolant system (RCS) pressure. The response will describe the related hardships associated with these alternative test options and finally describe how the Request I3R-11 test method would be effective in detecting potential piping leakage.

As outlined in Request I3R-11, with the reactor vessel closure head installed and the RCS depressurized, any pressurization of the annulus between the inner and outer O-rings creates pressure forces in a direction opposite to the design function of the inner O-ring, which could result in degradation or damage. Thus, that test option is not considered feasible.

Performing an alternative higher pressure test has two limitations. First, performing an alternative higher pressure test by pressurizing the leak-off piping at the beginning of the refueling outage while the reactor vessel closure head is still installed would require that RCS pressure always be higher than the O-ring annulus test pressure. Second, an alternative higher pressure test would not test the open-ended portion of line 075-BCB-3/4, the portion of the leak-off piping that originates outside of the outer O-ring and extends to normally closed valve BBV0079. The only test that does pressurize this portion of line 075-BCB-3/4 is the proposed test in Request I3R-11.

Alternative Higher Pressure Test Options:

It could be possible to connect a portable hydro pump skid at the blind flange connection down stream of valve BBV0081 to pressurize leak-off lines BB-076-BCB and BB-077-BCB through BBV0081 from the reactor vessel to the Code boundary isolation valve BBHV8032. This alternative higher pressure test could be performed at either of two operating plateaus: normal operating pressure and temperature in Mode 3 or at approximately 350 psia in Mode 5. Immediately after reactor shutdown for a refueling outage, the plant is stabilized in Mode 3, followed closely by RCS cool down. To perform a leak-off line alternative test at or near full RCS pressure would require holding the plant at normal operating pressure and temperature for

a minimum of six extra hours beyond the normal refueling outage sequence, while a portable hydro pump skid is installed and operated to pressurize and hold lines BB-076-BCB and BB-077-BCB at just below RCS operating pressure for a minimum of 4 hours prior to performance of the VT-2 examinations. At this time, the entire containment would have limited access as radiation protection personnel would not have sufficient time to survey and de-post the various areas required to setup and perform the test. Once a cool down starts, the plant enters Mode 5 approximately 6 hours later. The RCS is stabilized at approximately 350 psia to keep the reactor coolant pumps (RCPs) running for peroxide injection / crud removal (to minimize personnel dose during the refueling outage as a whole) and to cool down the RCS metal mass. This 350 psia plateau is maintained for about 12 hours. The portable hydro pump skid could be installed and operated at this lower pressure plateau. However, during the peroxide injection / crud removal process, which also occurs at this pressure level, the radiation levels increase such that radiation protection personnel prohibit entry into the areas required to setup and perform the test. As a result, performance of a leak-off line alternative test at this lower RCS pressure would require postponing peroxide injection / crud removal process for a minimum of six extra hours while a portable hydro pump skid is installed and operated to pressurize and hold lines BB-076-BCB and BB-077-BCB at just below current RCS system pressure for a minimum of 4 hours prior to performance of the VT-2 examinations.

Alternative Higher Pressure Test Method Hardships:

The performance of an alternative higher pressure test would impose the following hardships:

- The hydro pump test skid pressure and water quality must be adequately controlled, such that the reactor vessel O-rings are not over-pressurized or compromised. There is additional risk of O-ring damage and potential RCS leakage resulting from the alternative test method, as compared to the proposed Request I3R-11 test method.
- If an operational transient were to occur during testing such that the RCS pressure were to decrease below the O-ring annulus test pressure, inner O-ring loading would reverse resulting in a condition that has not been fully analyzed.
- Control Room operators would have to hold the RCS at an elevated pressure and temperature for a longer period (i.e., Mode 3 pressure test option) or postpone peroxide injection / crud removal process (i.e., Mode 5 pressure test option), either of which would impact critical path time at the start of the refueling outage.
- Additional personnel dose would be accrued to transport, install, test and remove the hydro pump test skid at a point in the refueling outage that would require continuous radiation protection personnel coverage (i.e., Mode 3 pressure test option).
- Since testing the reactor pressure vessel (RPV) leak-off piping prior to removing the reactor vessel closure head would involve the use of a hydro pump test skid, this could expose personnel stationed near pressurized vent or drain valves to unnecessary personal safety hazards in the event of a leak from or break of a non-class test pressure skid connection (i.e., Mode 3 pressure test option).

10 CFR 50.55a Request I3R-11 Test Method:

The above described alternative higher pressure test options are not consistent with the ASME Section XI Code intent for conduct of pressure tests in accordance with IWA-5000 and IWC-5000. In fact, ASME Code Case N-805, "Alternative to Class 1 Extended Boundary End of Interval or Class 2 System Leakage Testing of Reactor Vessel Head Flange O-ring Leak Detection System," was approved by the ASME Section XI Standards Committee and is included in the 2010 Edition, Supplement 6, of the ASME BPV Nuclear Code Case book. While Code Case N-805 has not yet been endorsed by the NRC in Regulatory Guide 1.147, "Inservice

Inspection Code Case Acceptability, ASME Section XI, Division 1," Request I3R-11 is consistent with the Code Case methods and is, therefore, an appropriate test approach.

The testing methodology proposed would be performed when the refueling pool is filled. During a refueling outage, the refueling pool is normally kept filled for several days while fuel assemblies are off-loaded and other outage activities are performed. The leak-off piping visual examination, as proposed in Request I3R-11, would be performed at some time after the refueling pool has been flooded for an extended period. As the visual examination is looking for through-wall flaws, the proposed test is considered a reasonable and sufficient method, even for small leak rates that result from the proposed low test pressure, since the piping is exposed to the pressure for a number of days. Additionally, as noted above, this test method is the only feasible test that tests line 075-BCB-3/4, the portion of the leak-off piping that originates outside of the outer O-ring (open ended) and extends to normally closed valve BBV0079.

Summary:

The test methodology of Request I3R-11 is appropriate for detecting through-wall leakage. Performance of either of the alternative higher pressure tests at the beginning of the refueling outage, prior to removing the reactor vessel closure head, is not justified based upon the additional operational risk and hardships. For the reason stated above, either of the alternative higher pressure tests would constitute a hardship without a compensating increase in the level of quality and safety.

2. *In Section 4.0 of the Attachment to the letter dated June 26, 2014, the licensee stated, in part,*

> *The installation of the plugs and subsequent use would incur additional radiological dose due to additional time for personnel at the reactor vessel flange.*

Please provide an estimate for amount of personnel radiation exposure (i.e., person-rem exposure) for the installation of the plugs and subsequent removal.

Response: Installation of pipe plugs has not been performed during the operational life of the plant. The condition of the threads in these two connections is unknown. Therefore, an accurate estimate of personnel radiation exposure for the installation of the plugs and subsequent removal cannot be made. However, based on past refueling outage dose surveys, the dose rates at the reactor vessel flange are 3 to 4 REM/hour. This information is provided to indicate the high radiation exposure for personnel attempting to install and remove these pipe plugs if such were attempted. The concerns documented in Request I3R-11 along with the potentially high dose received while attempting to install plugs, makes this option a significant hardship and an option that has no assurance of successfully testing the leak-off piping.

3. *Please explain if any segment(s) of the RPV flange leak-off piping is insulated and inaccessible for the ASME Code-required VT-2 visual examination. If yes, please discuss how the licensee will perform the VT -2 visual examination of the insulated and inaccessible segment(s) of the subject piping.*

Response: The length of the inner O-ring leak-off line BB-076-BCB is approximately 69 feet. The added length of line BB-077-BCB is approximately 6 feet. Line BB-077-BCB is insulated and accessible. Line BB-076-BCB is insulated and accessible from BB-077-BCB to the secondary shield wall, approximately 8 feet. The portion of line BB-076-BCB between the

secondary shield wall and the primary shield wall is not insulated and is accessible, approximately 28 feet. The remaining portion of BB-076-BCB runs through a primary shield wall reactor vessel main loop nozzle penetration and inside the reactor vessel main loop nozzle gallery between the inside of the primary shield wall and the reactor vessel. Portions of this piping run behind the primary shield wall mirror insulation on the inside of the primary shield wall. The entire piping in the nozzle gallery between the inside of the primary shield wall and the reactor vessel is inaccessible for direct VT-2 examination with the refueling cavity flooded. Line BB-075-BCB runs nearly parallel to BB-076-BCB with similar dimensions and configuration.

The examination of the insulated accessible piping will be in accordance with Section XI paragraph IWA-5242. The examination of the non-insulated accessible piping will be in accordance with Section XI paragraph IWA-5241(a). The examination of the inaccessible piping will be in accordance with Section XI paragraph IWA-5241(b) supplemented with subsequent VT-2 visual examination for boric acid residue indicative of leakage from the leak-off piping when the piping can be made accessible later in the refueling outage after drain down of the refueling cavity and access to the reactor vessel nozzle gallery is made available. This supplemental VT-2 visual examination will include opening of the mirror insulation that covers a portion of the inaccessible leak-off piping in the nozzle gallery to allow direct performance of the VT-2 visual examination.

4. *Please discuss the construction materials for the subject RPV flange leak-off piping. Discuss operating experience (e.g., plant-specific, fleet, and industry) regarding potential degradation of the subject piping due to any known degradation mechanisms that would lead to pipe through-wall leakage.*

Response: The RPV flange leak-off piping material is Type 304 Stainless Steel, Piping Class 2500#, ASME Section III Class 2.

In January 1988, during plant heat up following a refueling outage, WCNOC, identified leakage from the inner RPV flange O-ring seal. The plant was subsequently cooled down and the O-rings were replaced. No degradation in the RPV flange leak-off piping was identified. This was the only time the RPV leak-off piping at Wolf Creek Generating Station (WCGS) experienced elevated pressure since plant construction.

At Calvert Cliffs in January 1994, cracking was found in the RPV head O-ring leakage monitoring lines in both Units 1 and 2. Analysis of the failed material revealed that the lines cracked from transgranular stress corrosion cracking caused by elevated chloride concentration. It was concluded that the chlorides were present since construction.

WCNOC has not identified any degradation of the RPV flange leak-off piping during required VT-2 examinations at the end of each refueling outage, which includes accessible portions of the leak-off piping after pressurization from the static head of the flooded refueling cavity. In addition, in Refueling Outage 18 (Spring 2011), the mirror insulation was removed from the leak-off piping inside the reactor vessel main loop nozzle gallery (during walk down and work planning for reactor vessel nozzle mitigation) with no evidence of leak-off piping leakage identified.

5. *Please discuss the reactor coolant system leakage detection capabilities at the plant, or any measure(s) taken, to monitor and identify leakage in an unlikely event of a through wall leak in the RPV flange seal leak-off line piping concurrent with leak or failure of the RPV flange inner seal during normal operation.*

Response: During the operating cycle if the inner O-ring should leak it will be identified by an increase in temperature of the leak-off line above ambient temperature. This leak detection piping has temperature indication and a high temperature alarm in the Control Room, which is monitored by the Control Room Operators. This piping also acts as a leak-off line to collect leakage which is routed to the Reactor Coolant Drain Tank, whose level is indicated and alarmed in the Control Room. If inner O-ring leakage is identified, an operator is directed to enter containment and place the outer O-ring leakage detection in service by closing valve BBV0080 and opening valve BBV0079. The Control Room Operators then monitor outer O-ring seal for leakage. If outer O-ring seal leakage is identified, the Control Room Operators will initiate an orderly plant shutdown.

An unlikely event of a through wall leak in the RPV flange seal leak-off line piping concurrent with leak or failure of the RPV flange inner seal during normal operation would result in unidentified RCS leakage controlled by Technical Specification 3.4.13. Leakage detection systems have been designed to aid Control Room Operators in differentiating between possible sources of detected leakage within the containment and identifying the physical location of the leak. The RCS leakage detection system consists of the sump level and flow monitoring system, the containment air particulate monitoring system, the containment cooler condensate measuring system, and the containment humidity monitoring system. The sump level and flow monitoring system indicates leakage by monitoring increases in sump level. The containment cooler condensate measuring system and the containment humidity measuring system detect leakage from the release of steam or water to the containment atmosphere. The air particulate gas monitoring system detects leakage from the release of radioactive materials to the containment atmosphere.

The containment atmosphere particulate monitoring system provides the primary means of remotely determining the presence of reactor coolant leakage within the containment. Leakage to the containment atmosphere from the RCS would cause a change in the containment airborne radioactivity which would be detected by the air particulate monitors. Increases in containment airborne activity levels detected by either of the monitors indicate the reactor coolant pressure boundary as the source of leakage.

Since a leak in the flange leak-off piping concurrent with the inner O-ring leakage is a leak in the primary system, it would result in reactor coolant flowing into the containment normal or instrument tunnel sumps, which would be indicated by a level increase in the sumps. Indication of increasing sump level is transmitted from the sump to the control room level indicator by means of a sump level transmitter. The system provides measurements of low leakages by monitoring level increase versus time. Increases in the frequency of a particular containment sump pump operation or increases in the level in a particular sump facilitate localization of the source to components whose leakage would drain to that sump. If operator actions to place the outer O-ring seal leak-off line in service did not isolate the leak, operators would perform an RCS leak rate determination to identify the type of leak and quantify the leakage and take actions as required by Technical Specifications 3.4.13.

References:

1. WCNOC Letter ET 14-0019, "10 CFR 50.55a Request I3R-11 for the Third Inservice Inspection Program Interval," June 26, 2014. ADAMS Accession No. ML14182A091

2. Letter from C. F. Lyon, USNRC, to A. C. Heflin, WCNOC, "Wolf Creek Generating Station – Request for Additional Information Re: Request for Alternative I3R-11 (TAC No. MF4304)," July 25, 2014. ADAMS Accession No. ML14197A336