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

August 21, 2014

Mr. Adam C. Heflin
President, Chief Executive Officer,
and Chief Nuclear Officer
Wolf Creek Nuclear Operating Corporation
Post Office Box 411
Burlington, KS 66839

SUBJECT: WOLF CREEK GENERATING STATION – REQUEST FOR ADDITIONAL INFORMATION RE: REQUEST FOR ALTERNATIVE I3R-10 (TAC NO. MF4305)

Dear Mr. Heflin:

By letter dated June 26, 2014, Wolf Creek Nuclear Operating Corporation (the licensee) submitted to the U.S. Nuclear Regulatory Commission (NRC) for approval request for alternative I3R-10 for the Wolf Creek Generating Station. The licensee proposed an alternative to certain requirements of the American Society of Mechanical Engineers *Boiler and Pressure Vessel Code*, Section XI, IWB-5220, for the third 10-year interval of the inservice inspection program.

The NRC staff has reviewed the information provided in your application and determined that additional information is required in order to complete its formal review. The enclosed questions were provided to Mr. S. Wideman of your staff on August 18, 2014. Please provide a response to the enclosed questions within 45 days of the date of this letter.

If you have any questions, please contact me at 301-415-2296 or via e-mail at fred.lyon@nrc.gov.

Sincerely,

CFLyon

Carl F. Lyon, Project Manager
Plant Licensing Branch IV-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-482

Enclosure
Request for Additional Information

cc w/encl: Distribution via Listserv

REQUEST FOR ADDITIONAL INFORMATION

REQUEST FOR ALTERNATIVE I3R-10

TO ASME CODE REQUIREMENTS

WOLF CREEK NUCLEAR OPERATING CORPORATION

WOLF CREEK GENERATING STATION

DOCKET NO. 50-482

By letter dated June 26, 2014 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML14182A087), Wolf Creek Nuclear Operating Corporation (the licensee) submitted to the U.S. Nuclear Regulatory Commission (NRC) for approval request for alternative I3R-10 for the Wolf Creek Generating Station. The licensee proposed an alternative to certain requirements of the American Society of Mechanical Engineers *Boiler and Pressure Vessel Code* (ASME Code), Section XI, IWB-5220, for the third 10-year interval of the inservice inspection (ISI) program.

The NRC staff requests the following additional information in order to complete its review of the request.

1. For each pipe segment under consideration (i.e., Portion 1 through Portion 6 piping listed in Section 1.0 of Attachment to I3R-10), please provide the applicable examination category and item number as specified in Table IWB-2500-1, Section XI of the ASME Code.

2. Please clarify if any area(s) of Portion 1 through 6 piping in I3R-10 are insulated or inaccessible. If yes, discuss whether IWA-5241(b) and IWA-5242(b) will be followed. If these two subarticles will not be followed, discuss how the licensee will perform the VT-2 visual examination of the insulated or inaccessible area(s).

3. Please discuss whether there are any welded connections (e.g., butt or socket) in the pipe segments under consideration (i.e., Portion 1 through Portion 6 piping listed in Section 1.0 of Attachment to I3R-10). If yes, (a) discuss whether any of the welds have been examined by volumetric or surface examinations during the current 10-year ISI interval, and whether any weld(s) is in the risk-informed ISI program and has been or will be examined in the current 10-year ISI interval. (b) Discuss whether any pressure boundary leakage was identified during the current 10-year ISI interval in each pipe segment under consideration regardless of how the leakage was identified (e.g., from the ASME Code, Section XI, required pressure testing, boric acid corrosion control program walkdowns, reactor restart walkdowns, etc.).

4. Please discuss reactor coolant system (RCS) leakage detection capabilities at the plant, or any measure(s) taken, to monitor and identify leakage during operation in an unlikely event of a through-wall leak in the pipe segments under consideration.

Enclosure

5. As a basis for hardship for Portion 6 piping, the licensee stated, in part, that,

> Manually opening and closing these inboard valves at RCS pressure and temperature creates potential personnel safety issues.

   Please discuss whether hardship is due to Technical Specification limiting conditions, or a radiological dose (as low as is reasonably achievable (ALARA)) concern, or both. If ALARA is a concern, provide estimate for person roentgen equivalent man (rem).

6. For Portion 1 through 5 piping, please discuss whether the hardship is also due to a radiological dose (ALARA) concern. If yes, provide estimate for person roentgen equivalent man (rem).

August 21, 2014

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/RA/

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Plant Licensing Branch IV-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

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Request for Additional Information

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ADAMS Accession No.: ML14230A757 *email dated August 17, 2014

OFFICE	NRR/DORL/LPL4-1/PM	NRR/DORL/LPL4-1/LA	NRR/DE/EPNB/BC*	NRR/DORL/LPL4-1/BC (A)	NRR/DORL/LPL4-1/PM
NAME	FLyon	JBurkhardt	DAlley	EOesterle	FLyon
DATE	8/20/14	8/19/14	8/17/14	8/21/14	8/21/14

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