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JAFP-16-0114
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U.S. Nuclear Regulatory Commission
Document Control Desk
Washington, D. C. 20555

Subject: Response to Request for Additional Information for Proposed Inservice Inspection Program Alternative in Accordance with 10 CFR 50.55a(z)(2), RR-20 at James A. FitzPatrick Nuclear Power Plant

James A. FitzPatrick Nuclear Power Plant
Docket No. 50-333
License No. DPR-59

References:

1. ENOI letter, Proposed Inservice Inspection Program Alternative in Accordance with 10 CFR 50.55a(z)(2), RR-20 at James A. FitzPatrick Nuclear Power Plant, JAFP-16-0108, dated June 21, 2016
2. NRC memo, James A. FitzPatrick Nuclear Power Plant Inservice Inspection Program Alternative Relief Request RR-20 - Draft Request for Additional Information, dated June 24, 2016

Dear Sir or Madam:

James A. FitzPatrick Nuclear Power Plant (JAF) submitted a request on June 21, 2016, for an alternative to specific portions of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code, Section XI, “Rules for Inservice Inspection of Nuclear Power Plant Components,” pursuant to 10 Code of Federal Regulations (CFR) 50.55a(z)(2) [Reference 1].

The NRC made a request for additional information on June 24, 2016 [Reference 2]. The attachment to this letter contains the response to this request.

JAF requested approval of RR-20 [Reference 1] on or before June 28, 2016, to accommodate application of the request during the next potential outage. On June 24, 2016, due to an unrelated equipment issue, JAF has entered a shutdown. The Safety Relief Valve replacement activities have been planned for this shutdown and this will require a system leakage test and associated VT-2 visual examination of the mechanical joint connections during startup. Based on the current plant shutdown schedule, approval is requested by June 30, 2016.

There are no new regulatory commitments made in this letter. Should you have any questions, please contact the Regulatory Assurance Manager, Mr. William C. Drews, at (315) 349-6562.

Very truly yours,

William C. Drews
Regulatory Assurance Manager

WCD:dc

Attachment: Response to Request for Additional Information for RR-20

cc: USNRC, Regional Administrator, Region I
USNRC, Project Directorate
USNRC, Resident Inspector

JAFP-16-0114

Attachment

Response to Request for Additional Information for RR-20
(3 Pages)

Response to Request for Additional Information for RR-20

RAI-1

Page 2 of the relief request states that "…JAF may perform the required VT-2 leakage examination for any repair/replacement activities of mechanical joint connections performed in a future shutdown at a reactor pressure of ≥ 905 psig (consistent with 87% of the pressure required by IWB-5221(a) from ASME Code Case N-795)…

Please explain the statement "JAF may perform…" rather than "JAF will perform…

Response
The "JAF may perform…" statement on Page 2 of the Attachment for Relief Request RR-20 was referring to the time of a future shutdown. Upon approval, the relief request was meant to be an option during outage planning. During the emergent plant shutdown on June 24, 2016, James A. FitzPatrick (JAF) will perform replacement of 3-stage Safety Relief Valves (SRVs) with 2-stage SRVs. JAF will implement the Relief Request RR-20, if approved, for the current shutdown.

NRC approval of the Relief Request is sought prior to startup of the current outage in order to allow the performance of the VT-2 examinations on Class 1 SRV mechanical joint components at a pressure less than the Code-required pressure.

RAI-2

Page 3 of the relief request states that "…The proposed alternative provides an acceptable level of quality and safety and is consistent with requirements of ASME Code Case N-795 for a BWR Class 1 system leakage test, following repair/replacement activities. JAF proposes to perform the system leakage test at a pressure at least 5% (at least 55 psi) above the minimum pressure required by Code Case N-795." On page 2, the licensee stated that it may use a pressure of greater than 905 psig to perform the system leakage test.

The operating pressure at 100% rated power is 1040 psig. Therefore, 905 psig is 87% of 100% rated power (905/1040 = 87%). The above statement appears to indicate that JAF will perform the pressure test at 960 psig (905 psig + 55 psi). This discussion could be confusing when one area of the relief request proposes ≥ 905 psig and another area of the relief request proposes 960 psig. Please confirm that the system leakage test will be performed at a pressure of at least 960 psig and clarify as necessary.

Response
On the bottom of Page 2 at the start of the 'Proposed Alternative and Basis for Use' section, JAF was stating the minimum pressure of at least 905 psig to be consistent with Code Case N-795.

The maximum reactor pressure is associated with procedural limitations on Drywell entry due to plant conditions (15% power), at approximately 970 psig. The 960 psig test pressure is a pressure that is greater than 5% above the minimum test pressure of greater than or equal to 905 psig (905 psig being the minimum pressure required by Code Case N-795). The Page 3 reference to at least 55 psig above the minimum pressure required by the Code case is stating that JAF intends to perform the test at a higher pressure than minimum test pressure of 905 psig (namely at least 960 psig for added margin).

If RR-20 is approved, JAF would perform the system leakage test at a test pressure of at least 960 psig.

RAI-JAF-3

Page 2 of the relief request states in part that "Disposition of any observed leakage will consider the marginal increase in leakage rates that might occur at the nominal operating pressure associated with 100% rated reactor power (i.e. 1040 psig) and the actual reactor pressure when the examination was performed…" Please provide further explanation of this statement.

Response to Request for Additional Information for RR-20

Response
The statement attempted to recognize that Class 1 pressure boundary leakage rates detected at a lower approved Relief Request RR-20 test pressure would likely be greater if detected at the higher nominal operating pressure associated with 100% rated power (1040 psig). The recognition of higher leakage rates at higher pressures would be taken into consideration during evaluations for corrective action should leakage be detected. JAF will follow IWA-5250(a) and any deficiencies will be evaluated in the Corrective Action Program. In the case of identified leakage, a ratio comparing the lower test pressure to the higher operating pressure would be used during engineering evaluations of identified leakage.

RAI-JAF-4

If there are unplanned shutdowns with drywell entries before permanent cessation of operation, what additional action would the licensee take to inspect the affected mechanical joint connections to look for leakage?

Response
During startup from the current outage, after replacement of the SRVs, VT-2 examinations will be used to detect leakage from the affected mechanical joint connections.

During plant operation, drywell monitoring systems will be utilized to identify any leakage.

If there are additional unplanned shutdowns with drywell entries before permanent cessation of operation, JAF will perform an initial walk down upon entry to identify any evidence of leakage from SRV-related repair and replacement activities.

RAI-JAF-5

The NRC understands the licensee's desire to test at lower pressure. However, It is not clear to the NRC staff from the licensee's request how the needed pressure will be obtained. Based on other, similar requests, the NRC believes that the licensee may intend to use nuclear heat to obtain the necessary pressure.

a. Does the licensee propose to use nuclear heat to obtain the necessary pressure for testing?

b. If so, please discuss the risk to the plant associated with the proposed test method as compared with other available options. This discussion should include risks associated with the use of nuclear heat, as well as changes in plant alignment (including safety systems which must be removed from service) associated with the proposed and alternate methods of obtaining the necessary pressure. This discussion may be presented in tabular form if desired.

Response
a) Yes, if Relief Request RR-20 is approved, nuclear heat will be used to obtain the lower test pressure during the startup sequence.

b) During a typical post refuel outage system leakage test, the reactor vessel and main steam lines are essentially water solid and safety systems are isolated or in standby. The alternate method of using nuclear steam (heat) to obtain the required test pressure will use the Electro-Hydraulic Control (EHC) System to control pressure. This is a normal pressure control mode for a Boiling Water Reactor (BWR) with Reactor Pressure Vessel (RPV) temperature and level in more normal operating bands. Safety systems will be operable during this leakage test (ECCS, PCIS and RPS) as identified in the table below and there is no change to leakage indication available to monitor unidentified drywell leakage.

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Performance of a pressure test using nuclear steam is a safer method to pressure test a mechanical joint in a BWR than to perform a cold hydrostatic test for the following reasons:

a) A cold hydrostatic pressure test is performed with the RPV and steam lines flooded, which results in a nearly water solid condition. Pressure is controlled by carefully balancing RPV water injection with RPV water rejection/letdown. Small changes in this balance can create a rapid rate of pressure change. A cold hydrostatic test is performed at low temperature / high pressure. This condition places the plant significantly closer to the RPV pressure-temperature limits curve.

b) The nuclear steam pressure test is performed at high temperature/high pressure. This places the plant further from the RPV pressure-temperature (P-T) limits curve.

c) Fuel decay heat at the time of the proposed pressure test is much higher in this case than after a normal refueling outage cold hydrostatic test. More fuel decay heat impacts RPV pressure control.

Risk to the plant is minimized by performance of a leakage test using Relief Request RR-20 opposed to a cold hydrostatic leakage test because of significantly increased margin to RPV P-T limits curve with only a minimal reduction in test pressure; significantly more controllability of RPV pressure using the EHC system, and safety systems are available as opposed to steam-driven systems not being available during a cold hydrostatic test.

	System Leakage Test during startup with RR-20	**“Hydro” Test (ST-39H)**
High Pressure Coolant Injection	Operable	Isolated
Core Spray	Operable	Operable
Low Pressure Coolant Injection	Operable	Operable
Extensive valve manipulations	No (only per plant start-up)	Yes per ST-39H
Reactor Vessel and Main Steam lines are virtually water solid	No	Yes
RCIC	Operable	Isolated
Extensive water moves	No	Yes
Flood Risk	Steam will be present at a lower than operating pressure	Water at a higher pressure present therefore a greater flood risk
Energy input to the drywell if leakage is detected after replacement	More energy input due to higher steam temperature	Less energy input due to lower temperature water