NRR-PMDAPEm Resource

From:	HUFFMAN, WILLIAM C
Sent:	Friday, June 24, 2016 10:36 AM
To:	Bill Drews (WDrews1@entergy.com); 'Hawes, Mark'
Cc:	TSAO, JOHN C; ALLEY, DAVID W; ANDERSON, SHAUN M; WENGERT, THOMAS J; LAMB, JOHN G
Subject:	James A. FitzPatrick Nuclear Power Plant Inservice Inspection Program Alternative Relief Request RR-20 - Draft Request for Additional Information

Mr. William C. Drews
Regulatory Assurance Manager
Entergy Nuclear Operations, Inc.
James A. Fitzpatrick Nuclear Power Plant

REQUEST FOR ADDITIONAL INFORMATION

REGARDING PROPOSED INSERVICE INSPECTION PROGRAM ALTERNATIVE

IN ACCORDANCE WITH 10 CFR 50.55a(z)(2), RR-20

ENTERGY NUCLEAR OPERATIONS. INC.

JAMES A. FITZPATRICK NUCLEAR POWER PLANT

DOCKET NO. 50-333

RENEWED FACILITY OPERATING LICENSE NO. DPR-59

By letter dated June 21, 2016 (Agencywide Documents and Access Management System Accession No. ML16173A465; CAC MF8013), Entergy Nuclear Operations, Inc. (the licensee) requested relief from the requirements of American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code, Section XI, IWB-5221 for James A. FitzPatrick Nuclear Power Plant. The licensee proposed to use Relief Request RR-20 to perform a system leakage test of Class 1 components at a lower pressure than the ASME Code required pressure.

The licensee is anticipating the possible need for a short duration maintenance shutdown to replace certain main steam safety relief valves. The requested proposed alternative would allow the performance of the VT-2 visual leakage examination following main steam safety relief valve repair and replacement activities at the lower pressure of greater than or equal to 905 psig while employing a one hour hold time for non-insulated components and a six hour hold time for insulated components. The licensee states that performance of a cold leakage test (that is, a non-nuclear heat-up such as that required following a refueling outage) subsequent to a maintenance shutdown is judged to be an imprudent course of action for the reasons identified in its submittal. The licensee states that the alternative is therefore justified since compliance with the cited requirements of the identified code would result in a plant hardship without a commensurate increase in the level of quality and safety of the associated maintenance activity.

The NRC staff has determined that additional information is needed to continue the review as discussed below.

This request for additional information (RAI) is identified as draft at this time to confirm your understanding of the RAI and of the information needed to complete our evaluation. If the request for information is understood, please respond to this RAI by June 27, 2016, in order to support your requested action date of June 28, 2016. Otherwise, please schedule a clarification call as soon as reasonably possible.

Please call me at 301-415-2046 if you would like to set up a conference call to clarify the request for information.

Respectfully,

Bill Huffman
Project Manager
NRR/DORL/LPL4-2
U.S. Nuclear Regulatory Commission
William.Huffman@nrc.gov

RAI-JAF-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…"

RAI-JAF-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.

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.

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?

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.

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