Resource, NRR-PMDAPEm

From: KLOS, John J
Sent: Thursday, June 23, 2016 8:47 AM
To: Williams, Lisa L.
Cc: KLOS, John J
Subject: RAIs: Columbia MF7154 Relief Request 4ISI-02, Use of Code Case N-795 following repair replacement activities, 30 day response, due Friday July 22 2016

Ms. Williams,

By letter dated December 10 2015, Agencywide Documents Access and Management System (ADAMS) Accession No. ML15344A519, Energy Northwest (the licensee) submitted a relief request related to the pressure test of Class 1 components and proposed an alternative using provisions of American Society of Mechanical Engineers (ASME) Code Case N-795 due to a hardship condition.

The Nuclear Regulatory Commission (NRC) staff has reviewed the submittal and has determined that requests for additional information (RAIs) are needed to complete its technical review and make a regulatory finding regarding this relief request. The draft questions were sent via electronic transmission on June 10, 2016 and a clarification call was held on June 20, 2016.
Additionally, it was agreed that a response would be submitted within 30 calendar days from the date of this email, on July 22, 2016.

REQUEST FOR ADDITIONAL INFORMATION
RELIEF REQUEST 4ISI-02 REGARDING PRESSURE TEST FOLLOWING REPAIR/REPLACEMENT ACTIVITIES
ENERGY NORTHWEST
COLUMBIA GENERATING STATION
DOCKET NUMBER 50-397

Background:

By letter dated December 10, 2015, Agencywide Documents Access and Management System (ADAMS), Accession Number ML15344A519, Energy Northwest (the licensee) requested relief from the requirements of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (B&PV Code) specifically related to the pressure test of the Class 1 components in the containment, excluding the reactor pressure vessel (RPV), following repair/replacement activities prior to return the plant to service at the Columbia Generating Station (Columbia). In relief request 4ISI-02, an alternative pressure test is proposed that implements provisions that are similar to ASME Code Case N-795 “Alternative Requirements for boiling water reactor (BWR) Class 1 System Leakage Test Pressure Following Repair/Replacement Activities, Section XI, Division 1”.

Regulatory Basis:

Pursuant to Title 10 of the *Code* of *Federal Regulations* (10 CFR), Part 50, paragraph 55a(g)(4), Inservice Inspection Requirements, ASME Code Class 1, 2, and 3 components (including supports) shall meet the requirements, except the design and access provisions and the preservice examination requirements, set forth in the ASME Code, Section XI, "Rules for Inservice Inspection of Nuclear Power Plant Components," to the extent practical within the limitations of design, geometry, and materials of construction of the components. The regulations require that inservice examination of components and system pressure tests conducted during the first 10-year inspection interval and subsequent 10-year inspection intervals comply with the requirements in the latest edition and addenda of Section XI of the ASME Code incorporated by reference in 10 CFR

50.55a(b), 12 months prior to the start of the 120-month inspection interval, subject to the limitations and modifications listed therein.

Paragraph 55a(a)(3) of 10 CFR Part 50 states, in part, that alternatives to the requirements of 10 CFR 50.55a(g) may be used, when authorized by the NRC, if (i) the proposed alternatives would provide an acceptable level of quality and safety or (ii) compliance with the specified requirements would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety.

Pursuant to Title 10 of the Federal Code of Regulations (10 CFR) 50.55a(b)(2)(xxvi), Section XI condition: Pressure testing Class 1, 2, and 3 mechanical joints. The repair and replacement activity provisions in IWA-4540(c) of the 1998 Edition of Section XI for pressure testing Class 1, 2, and 3 mechanical joints must be applied when using the 2001 Edition through the latest edition and addenda incorporated by reference in paragraph (a)(1)(ii) of this section.

Requests:
To complete its review, the U. S. Nuclear Regulatory Commission (NRC) staff requests the following additional information.

1. Clarify that relief request 4ISI-02 is applicable, or used, only after completion of the Table IWB-2500-1 (Examination Category B-P) required system leakage testing. (In other words, the proposed alternative pressure test shall not be used to satisfy the requirements in Table IWB-2500-1 (Examination Category B-P).

2. The licensee states that the scope of the relief request is ASME Section III Class 1 system components excluding the reactor vessel with specific hold times applied to non-insulated and insulated components (per Section 1.0 and 4.0 of the relief request). The licensee also states in Section 4.0 that

   "The use of Code Case N-795 following repair/replacement activities at Columbia would allow execution of system leakage tests and VT-2 visual examinations during normal plant startup conditions at low reactor power levels for some Class 1 repair/replacement activities located in containment."

   Also pursuant to Title 10 of the *Federal Code of Regulations* (10 CFR) 50.55a(b)(2)(xxvi), Section XI Condition: Pressure testing Class 1, 2, and 3 mechanical joints; The repair and replacement activity provisions in IWA-4540(c) of the 1998 Edition of Section XI for pressure testing Class 1, 2, and 3 mechanical joints must be applied when using the 2001 Edition through the latest edition and addenda incorporated by reference in paragraph (a)(1)(ii) of this section.

   Therefore, it is not clear to the NRC staff the scope of the licensee's relief request following repair/replacement activities prior to return the plant to service.

   Please clarify whether this relief request's scope includes only isolable, or non-isolable components or whether it includes both. Additionally please describe how the mechanical joints will be subjected to this relief's request pressure testing.

3. ASME Code Case N-795 allows an alternative lower test pressure for Class 1 pressure tests following repair/replacement activities at BWR nuclear power plants using a critical reactor core to raise the temperature of the reactor coolant and pressurize the reactor coolant pressure boundary (RCPB). This code case was developed because a majority of BWR plants must perform a pressure test that requires the primary system to obtain a test pressure corresponding to 100 percent rated reactor power and allow access for the examination. During this test, the vessel is filled essentially water solid while at a greatly reduced margin to cold overpressure conditions. Licensees have asserted that performance of the primary system pressure test under these conditions places the unit in a position of significantly reduced margin, approaching the fracture toughness limits defined in the Technical Specification Pressure-Temperature curves. This is because the pressure control system does not allow the setpoint

to approach the 100 percent pressure value. Also, the core reload analysis does not cover the elevated pressure at low power levels conducive to personnel entry into the drywell.

The NRC has a long-standing prohibition against the production of heat through the use of a critical reactor core to raise the temperature of the reactor coolant and pressurize the RCPB.

This position is documented in the following references;

a) Dated February 2, 1990, a letter from James M. Taylor, NRC Executive Director for Operations to Messrs. Nicholas S. Reynolds and Daniel F. Stenger, Nuclear Utility Backfitting and Reform Group, ADAMS Accession No. ML14273A002, established the NRC's position with respect to use of a critical reactor core to raise the temperature of the reactor coolant and pressurize the RCPB.

b) Information Notice (IN) 98-13, "Post-Refueling Outage Reactor Pressure Vessel Leakage Testing before Core Criticality," dated April 20, 1998, where a licensee had conducted an ASME Code, Section XI, leakage test of the reactor pressure vessel and subsequently discovered that it had violated 10 CFR part 50, Appendix G, to complete pressure and leak testing before the core was taken critical. The Information Notice reiterates the NRC's position that under the ASME Code, Section XI, Class 1 and 2 leakage tests provide a level of defense-in-depth for detecting pressure boundary leakage. From a safety perspective, performing this test using nuclear heat defeats the intended purpose of ensuring the integrity of the RPV as a fission product barrier.

c) A final rule published in the Federal Register on December 19, 1995, clarified the NRC staff position in 10 CFR part 50, Appendix G, Section IV.A.2.d, as follows: "Pressure tests and leak tests of the reactor vessel that are required by Section XI of the ASME Code must be completed before the core is critical."

These references form the bases for the NRC's position concerning this issue which are as follows:

a) The RCPB is one of the principal fission product barriers. In accordance with the defense-in-depth safety precept, nuclear power plant design provides multiple barriers to the accidental release of fission products from the reactor. Additionally, nuclear operation of a plant should not commence before completion of system hydrostatic and leakage testing to verify the basic integrity of the RCPB, a principal defense-in-depth barrier to the accidental release of fission products. The assured integrity of the RCPB, and adequacy of these inspections, is fundamental to the safe operation of nuclear power plants and is, therefore, of critical importance in adequately assuring the protection of public health and safety. For this reason, General Design Criteria 14 requires explicitly that the RCPB be designed, fabricated, erected, and tested so as to have an extremely low probability of abnormal leakage, of rapidly propagating failure, and of gross rupture. Consistent with this conservative approach to the protection of public health and safety, and the critical importance of the RCPB in preventing accidental release of fission products, the NRC has always maintained the view that verification of the integrity of the RCPB is a necessary prerequisite to any nuclear operation of the reactor. Initiation of criticality for the purpose of hydro testing or leakage testing to verify RCPB integrity is contrary to this basic safety principle.

b) The NRC's view has been that hydro testing must be done essentially water solid so that stored energy in the reactor coolant is minimized during a hydrotest or leak test.

c) The initiation of criticality creates a severe working environment that encumber required inspections to such an extent as to call into serious question the adequacy and ability of those inspections to properly verify reactor coolant boundary integrity. The elevated reactor coolant temperatures result in a severely uncomfortable and difficult working environment in plant spaces where the system leakage inspections must be conducted. The greatly increased stored energy in the reactor coolant increases the hazard to personnel and equipment in the event of a leak, and the elevated temperatures contribute to increased concerns for personnel safety due to burn hazards, even if there is no leakage. As a result, the ability for plant workers to perform a comprehensive and careful inspection becomes greatly diminished.

In summary, the NRC's position is that testing under these conditions involves serious impediments to careful and complete inspections, and thus, inherent uncertainty with regard to assuring the integrity of the RCPB. Further, the practice is not consistent with basic defense-in-depth safety principles.

However, as evidenced by the precedent cited in this relief request 4ISI-02, the NRC has authorized pressure tests at a number of plants on the basis that compliance with the specified requirements would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety.

In this relief request, the licensee proposed alternative pressure testing in accordance with 10 CFR 50.55a(z)(2) "The licensee must demonstrate compliance with the specified requirements of this section would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety."

Please describe the method(s) and procedures that can be used for attaining 100 percent normal operating pressure required by IWB-5221(a) in order to perform the Code-compliant system leakage test and specifically describe the hardship or unusual difficulty associated with each method. This discussion should be similar to the procedure currently performed during refueling outages, but please specify what the risks are and include a discussion of the following items;

a) Which equipment would be available during the Code-compliant pressure testing;
b) Whether there would be changes to the shutdown interlocks;
c) The abnormal plant conditions/alignments necessary to complete this testing;
d) The available sources for heat removal and pressure/inventory control and whether they would be sufficient for the purposes of this test following a maintenance prior to return the plant to service.

4. Describe the method for attaining and holding 87 percent of normal operating pressure in order to perform the proposed alternative leakage test and explain why pressurization to 87 percent of normal operating pressure, with a hold for 8 hours for insulated components and one hour for non-insulated components, is possible but pressurizing to 100 percent of normal operating pressure is unusually difficult.

   Given that the proposed alternative will use nuclear heat to generate the necessary pressure needed to perform the repair/replacement inspections, the NRC staff needs additional information to confirm that the proposed alternative will provide reasonable assurance of structural integrity and leak tightness. Please provide the following information for cases where the post repair/replacement leak test is performed at the reduced pressure permitted by ASME Code Case N-795. Provide the information for two cases: 1) Using nuclear heat and 2) not using nuclear heat. For both cases, please provide the following information in a tabular format:

   a) Available methods and systems for heat removal (including heat removal capacity for each available system);
   b) Available methods and systems for pressure control;

c) Available methods and systems for inventory control;
d) Available methods and systems for reactivity control;
e) An indication of the mode and plant operating state;
f) Any changes to normal interlocks;
g) An indication of whether the heat removal, pressure, inventory, and reactivity control would be sufficient for the configuration of the plant.

5. The NRC staff understands that the proposed alternative will use nuclear heat to generate the necessary test pressure needed to perform the repair/replacement inspections. However, these pressure tests are necessary to ensure the integrity of the RCPB. Discuss the Columbia operating experience associated with testing the integrity of repairs/replacements on the RCPB, associated with repair/replacement activities which would fall in the scope of this request. Include the type of repair/replacement activity (mechanical or welded), NPS size of the component, whether the component was isolable/non-isolable, the method used to obtain the required test pressure and VT-2 examination, and the results of the VT-2 examination.

John Klos
DORL Callaway, Columbia Project Manager
U.S. NRC, Office of Nuclear Reactor Regulation,
Division of Operating Reactor Licensing, O8E7
NRC/NRR/DORL/LPL4-1, MS O8H4A
Washington, DC 20555-0001
301.415.5136, 301.415.2102 (fax)
John.Klos@NRC.gov

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Created By: John.Klos@nrc.gov

Recipients:
"KLOS, John J" <John.Klos@nrc.gov>
Tracking Status: None
"Williams, Lisa L." <llwilliams@energy-northwest.com>
Tracking Status: None

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