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

October 31, 2016

Mr. Mark E. Reddemann
Chief Executive Officer
Energy Northwest
P.O. Box 968 (Mail Drop 1023)
Richland, WA 99352-0968

SUBJECT: COLUMBIA GENERATING STATION - REQUEST FOR ALTERNATIVE 4ISI-02 APPLICABLE TO THE FOURTH 10-YEAR INSERVICE INSPECTION PROGRAM INTERVAL (CAC NO. MF7154)

Dear Mr. Reddemann:

By letter dated December 10, 2015, as supplemented by letter dated July 14, 2016, Energy Northwest (the licensee) submitted a request to the U.S. Nuclear Regulatory Commission (NRC) for an alternative from the requirements of certain American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code), specifically related to the pressure test of the Class 1 pressure-retaining components following repair/replacement activities in an unscheduled maintenance outage event or a forced outage event prior to returning the plant to service at the Columbia Generating Station (CGS). In this request, 4ISI-02, the licensee requested the use of an alternative to pressure testing requirements after repair/replacement activities. The proposed provisions are similar to ASME Code Case N-795, "Alternative Requirements for BWR [Boiling-Water Reactor] Class 1 System Leakage Test Pressure Following Repair/Replacement Activities, Section XI, Division 1." The ASME Code Case N-795 has not been approved for use by the NRC staff in Regulatory Guide 1.147, Revision 17.

Specifically, pursuant to Title 10 of the *Code of Federal Regulations* (10 CFR) 50.55a(z)(2), the licensee requested to use an alternative pressure test following repair/replacement activities prior to returning the plant to service on the basis that complying with the specified requirement would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety.

The NRC staff has reviewed the subject request and concludes, as set forth in the enclosed safety evaluation, that Energy Northwest has adequately addressed all of the regulatory requirements set forth in 10 CFR 50.55a(z)(2). Therefore, the NRC staff authorizes the use of the licensee's proposed alternative at CGS for the fourth 10-year inservice inspection interval, which started on December 13, 2015, and is scheduled to end on December 12, 2025.

The authorization of the proposed alternative in relief request 4ISI-02 does not imply or infer the NRC approval of ASME Code Case N-795. All other ASME Code, Section XI, requirements for which relief was not specifically requested and approved herein by the NRC staff remain applicable, including third-party review by the Authorized Nuclear Inservice Inspector.

If you have any questions regarding this matter, please contact the NRC project manager, John Klos, at (301) 415-5136 or via e-mail at John.Klos@nrc.gov.

Sincerely,

Robert J. Pascarelli, Chief
Plant Licensing Branch IV-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-397

Enclosure:
Safety Evaluation

cc w/encl: Distribution via Listserv

UNITED STATES
NUCLEAR REGULATORY COMMISSION
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SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

REQUEST FOR ALTERNATIVE 4ISI-02 REGARDING PRESSURE TEST FOLLOWING REPAIR/REPLACEMENT ACTIVITIES PRIOR TO RETURNING PLANT TO SERVICE

ENERGY NORTHWEST

COLUMBIA GENERATING STATION

DOCKET NO. 50-397

1.0 INTRODUCTION

By letter dated December 10, 2015 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML15344A519), as supplemented by letter dated July 14, 2016 (ADAMS Accession No. ML16196A418), Energy Northwest (the licensee) requested an alternative from the requirements of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code), specifically related to the pressure test of the Class 1 pressure-retaining components following repair/replacement activities in an unscheduled maintenance outage event or a forced outage event prior to returning the plant to service at Columbia Generating Station (CGS). In this request, 4ISI-02, the licensee proposes an alternative to pressure testing requirements after repair/replacement activities. The proposed provisions are similar to ASME Code Case N-795, "Alternative Requirements for BWR [Boiling-Water Reactor] Class 1 System Leakage Test Pressure Following Repair/Replacement Activities, Section XI, Division 1." The ASME Code Case N-795 has not been approved for use in NRC Regulatory Guide (RG) 1.147, Revision 17, "Inservice Inspection Code Case Acceptability, ASME Section XI, Division 1," August 2014 (ADAMS Accession No. ML13339A689).

Specifically, pursuant to Title 10 of the *Code of Federal Regulations* (10 CFR) 50.55a(z)(2), the licensee requested to use the proposed alternative on the basis that complying with the specified requirement would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety.

2.0 REGULATORY EVALUATION

In accordance with 10 CFR 50.55a(g)(4), *Inservice inspection standards requirement for operating plants*, throughout the service of life of a boiling or pressurized water-cooled nuclear power facility, ASME Code Class 1, Class 2, and Class 3 components (including supports) must be in compliance with the requirements in the applicable edition and addenda of Section XI of the ASME Code, except design and access provisions and preservice examination

Enclosure

requirements. The requirements are set forth in Section XI of editions and addenda of the ASME Code that become effective subsequent to editions specified in paragraphs (g)(2) and (3) of 10 CFR 50.55a and that are incorporated by reference in paragraph (a)(1)(ii) of 10 CFR 50.55a, to the extent practical within the limitations of design, geometry, and materials of construction of the components.

Pursuant to 10 CFR 50.55a(g)(4)(ii), *Applicable ISI [inservice inspection] Code: Successive 120-month intervals*, inservice examination of components and system pressure tests conducted during successive 120-month inspection intervals must comply with the requirements of the latest edition and addenda of the Code incorporated by reference in paragraph (a) of 10 CFR 50.55a 12 months before the start of the 120-month inspection interval (or the optional ASME Code Cases listed in RG 1.147, Revision 17), when using Section XI, that are incorporated by reference in paragraphs (a)(3)(ii) of 10 CFR 50.55a, subject to the conditions listed in paragraph (b) of 10 CFR 50.55a. However, a licensee whose ISI interval commences during the 12- through 18-month period after July 21, 2011, may delay the update of their ASME Code, Section XI, Appendix VIII program by up to 18 months after July 21, 2011.

Pursuant to 10 CFR 50.55a(z), *Alternatives to codes and standards requirements*, alternatives to the requirements of paragraphs (b) through (h) of 10 CFR 50.55a or portions thereof may be used when authorized by the Director of the Office of Nuclear Reactor Regulation or the Office of New Reactors, as appropriate. A proposed alternative must be submitted and authorized prior to implementation. The applicant or licensee must demonstrate that: (1) acceptable level of quality and safety, meaning the proposed alternative would provide an acceptable level of quality and safety; or (2) hardship without a compensating increase in quality and safety, meaning compliance with the specified requirements of 10 CFR 50.55a would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety.

Pursuant to 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 ASME Code, 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 10 CFR 50.55a.

Based on the analysis of the regulatory requirements, the NRC staff concludes that the regulatory authority exists to authorize the licensee's proposed alternative 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.

3.0 TECHNICAL EVALUATION

3.1 The Licensee's Request for Alternative

3.1.1 Background

By letter dated August 9, 2013 (ADAMS Accession No. ML13191A054), the NRC authorized the same alternative pressure test of Class 1 components, excluding the reactor pressure vessel (RPV), following repair/replacement activities prior to returning the plant to service for the third 10-year ISI interval at CGS.

3.1.2 Component Affected

The affected components are ASME Code Class 1 pressure-retaining components in containment, excluding the RPV. The licensee stated that scope of this relief request includes:

- Non-isolable and isolable non-welded mechanical joints repair/replacement activities on components such as main steam relief valves, control rod drives (CRD), reactor recirculation (RRC) pump mechanical seals, and in-core instrumentation

- Non-isolable and isolable welded connections repair/replacement activities

The licensee stated that the proposed reduced test pressure will not be used to satisfy the requirements for pressure test of the RPV. The licensee also stated that the proposed reduced test pressure will not be used to satisfy the requirements of Table IWB-2500-1, Examination Category B-P, of Section XI in the ASME Code.

3.1.3 Applicable Code Edition and Addenda

The Code of record for the fourth 10-year ISI interval is the 2007 Edition through 2008 Addenda of the ASME Code.

3.1.4 Duration of Relief Request

The licensee submitted this request for an alternative for the fourth 10-year ISI interval, which started on December 13, 2015, and is scheduled to end on December 12, 2025.

3.1.5 ASME Code Requirement

The ASME Code requirements applicable to this request originate in Section XI, IWA-4000, IWA-5000, and IWB-5000.

- The requirements in IWA-4540(a) state that the repair/replacement activities performed by welding or brazing on a pressure-retaining boundary shall include a hydrostatic or system leakage test in accordance with IWA-5000, prior to, or as part of, returning to service.

- In accordance with IWA-5212(a), system leakage tests and system hydrostatic tests shall be conducted at the pressure and temperature specified in IWB-5000, IWC-5000, and IWD-5000.

- In accordance with IWB-5221(a), the system leakage test shall be conducted at a pressure not less than the pressure corresponding to 100 percent rated reactor power.

- In accordance with IWA-5213(b), a 10-minute holding time for non-insulated components, or a 4-hour holding time for insulated components, is required after attaining test pressure.

The requirements specific to Class 1, 2, and 3 mechanical joints originate in IWA-4540(c) of Section XI to the 1998 Edition of the ASME Code, as mandated by 10 CFR 50.55a(b)(2)(xxvi).

- In accordance with IWA-4540(c) of the 1998 Edition, mechanical joints made in installation of pressure-retaining items shall be pressure tested in accordance with IWA-5211(a). Mechanical joints for component connections, piping, tubing (except heat exchanger tubing), valves, and fittings, nominal pipe size (NPS)-1 and smaller, are exempt from the pressure test.

3.1.6 <u>Proposed Alternative</u>

The licensee proposed an alternative to perform the pressure test of the Class 1 components identified in this request at a reduced test pressure following repair/replacement activities in an unscheduled maintenance outage event or a forced outage event prior to return the plant to service. Specifically, the licensee's proposed alternative is to implement provisions that are similar to ASME Code Case N-795. This code case has not been incorporated by reference into 10 CFR 50.55a by inclusion in RG 1.147, Revision 17.

In its letters dated December 10, 2015, and July 14, 2016, the specifics of the licensee's proposed alternative pressure test are outlined as follows:

- The licensee proposed to use, at a minimum, the ASME Code Case N-795 specified test pressure (i.e., at least 87 percent of the IWB-5221(a) required pressure) to conduct pressure testing.

- The licensee stated that the nominal operating pressure at 100 percent rated reactor power at CGS is 1020 pounds per square inch gauge (psig), thus, the proposed test pressure will be at least 888 psig (i.e., at least 87 percent of 1020 psig). The licensee will obtain the proposed test pressure by use of nuclear heat during normal operational startup sequence.

- The licensee proposed to use a longer hold time than the hold time specified in Code Case N-795. The licensee's proposed extended hold time is 1 hour for non-insulated components and 8 hours for insulated components prior to performing the ASME Code-required VT-2 visual examinations following pressurization of components for pressure testing.

The licensee stated that it will not use the proposed reduced test pressure to satisfy the requirements for pressure test of the RPV or to satisfy the requirements in Table IWB-2500-1, Category B-P.

3.1.7 <u>Basis for Alternative</u>

In its letter dated July 14, 2016, the licensee discussed the nuclear safety aspect of the pressure test performed at pressure of at least 888 psig as part of a normal startup evolution. In the normal operational startup sequence and procedure, all systems are in normal alignment and all core standby cooling systems are available. The licensee stated that at 5 percent of reactor power, the RPV level is controlled by the feedwater level control system with reject flow to the reactor water clean-up (RWCU) system. Pressure would be controlled by the digital electrohydraulic (DEH) control system with steam being routed by the turbine bypass valves to the condenser. Normal systems for reactivity control such as control rods would be available. The licensee also stated that no shutdown interlocks would be affected. The plant would be in Mode 2 with the reactor mode switch in Startup/Hot Standby and the plant would be in a configuration with normal pressure, inventory, and reactivity control. This permits safe access to drywell for the component pressure test and the associated VT-2 visual examination.

The license stated that the proposed hold time, which is longer than the ASME Code-required hold time, provides additional time for leakage to be identified during the associated VT-2 visual examination following pressurization at the proposed test pressure of 888 psig.

However, the license stated that the ASME Code pressure test at the normal operating pressure of 1020 psig after repair/replacement activities requires abnormal plant conditions and system alignments which create significant operational challenges, and are accompanied with inherent risk that may affect safe operation of the plant. In its letter dated July 14, 2016, the licensee tabulated the nuclear safety and risks associated with performance of a pressure test by nuclear heat (Case 1) as compared to performance of pressure test by non-nuclear heat (Case 2) as follows.

	Case 1: Nuclear Heat	**Case 2: Non-Nuclear Heat**
Method	Pressure test at 87 percent of normal operating pressure during normal operational startup sequence	Pressure test at 100 percent of normal operating pressure
RPV and Main steam lines	Not water-solid	Essentially water-solid
Heat removal	By directing steam to the main condenser hotwell	Using residual heat removal (RHR) shutdown cooling (isolated at 48 psig); and RWCU via non-regenerative heat exchanger
Pressure control	DEH system	Using CRD system and RWCU system along with main steam drain orifice bypass valve if main condenser is available
Inventory (level) control	Using Feedwater system with RWCU reject flow	Using CRD system with letdown to the RWCU system
Reactivity control	Using control rods	Control rods all in

	Case 1: Nuclear Heat	Case 2: Non-Nuclear Heat
Mode	Mode 2 (Startup/Hot Standby)	Mode 4
Normal interlock	All active	Interlocks defeated (Reactor Protection System (RPS) Main Steam Isolation Valve (MSIV) scram interlock and RPS input for RPV Steam Dome Pressure; and bypassed MSIV closure on low condenser vacuum)
RPV pressure-temperature (P-T) limits	Not affected	Approaching Technical Specification (TS) P-T limits

The licensee stated that the pressure testing by nuclear heat at the proposed pressure of 888 psig during normal reactor startup conditions with all systems in normal alignment and all core cooling system available would significantly minimize risk to safe operation of the plant and personnel exposure to radiation when compared to the pressure testing by non-nuclear heat.

3.1.8 Basis for Hardship

In its letter dated July 14, 2016, the licensee discussed the hardship and unusual difficulty associated with risk and safety of plant operation if the Code-compliant pressure test at 100 percent rated power is performed. A summary is provided below:

1. Nuclear heat method

   The licensee stated that to accommodate for Code-compliant pressure testing, nuclear heat could be used to raise the primary system pressure to the normal operating pressure of 1020 psig during plant startup. For the nuclear heat method, the licensee considered the following plant conditions.

   a. The licensee stated that the operating pressure of 1020 psig can be attained using nuclear heat at 100 percent rated power during normal startup procedures and power ascension. At 100 percent rated power, access to containment is not permitted. Personnel that need to enter containment to perform the associated VT-2 visual examinations following pressurization would be exposed to high radiation dose rates. Personnel exposure to high radiation is contrary to the as low as reasonably achievable (ALARA) criteria. In addition, the containment entry at 100 percent rated power would result in de-inerting the containment, which creates an abnormal plant condition and system lineups that would result in operational risks.

   b. The licensee stated that attaining the normal operating pressure of 1020 psig during startup at 5 percent reactor power would require a significant change to the steam pressure control system by means of manipulating valves and controls. Although it is technically feasible to manipulate these controls to achieve the nominal system pressure of

1020 psig at this reactor power, the licensee has neither performed a nuclear analysis to support this mode of operation and plant condition nor attempted to manipulate the control systems to establish the normal operating pressure at this plant condition. At this plant condition, containment entry is permitted. Attempting to establish nominal operating pressure of 1020 psig during plant startup requires changing the setpoints outside the normal range of operation and affects core reactivity. The lack of experience and predictability of setting the pressure regulators outside the normal range of operation challenges the plant operations with potential risk of adversely impacting the reactor safety.

2. Non-nuclear heat method

The licensee stated that the Code-compliant pressure testing could be accomplished by non-nuclear heat methods before startup. For the non-nuclear heat method, the licensee considered the following to attain the normal operating pressure of 1020 psig for Code-compliant pressure testing.

a. The licensee stated that the decay heat and RRC pump heat could be used to heat up and pressurize the primary systems to the normal operating pressure of 1020 psig. For this method, the RPV would be filled with coolant and the steam lines would be flooded to the outboard MSIV to provide an essentially water-solid condition. Electrical jumpers to bypass MSIV valves and system check valves would be required to accommodate pressure test. The higher decay heat would create a significant challenge to the plant operations staff while performing pressurization for the test during a forced maintenance outage of a short duration. Thus, the RHR shutdown cooling would be required to be removed from service. The isolation of the RHR shutdown cooling under high decay heat loads creates abnormal plant conditions and system alignments, and is accompanied with inherent risk. Extensive valve manipulations, system lineups, and procedural controls would be required to establish the test pressure of 1020 psig during outage conditions without withdrawal of the control rods and while maintaining the reactor coolant system (RCS) pressure, temperature, heat-up, and cooldown rate within TS limits. Therefore, significant operational challenges and abnormal plant conditions would create a more error-likely environment that could adversely impact the reactor safety.

b. The licensee stated that the decay heat could be used to produce sufficient pressure to conduct the pressure test at normal operating pressure of 1020 psig while maintaining the RPV level at its normal operational level. Since this method is a modification of the above method in 2.a, the same challenges and abnormal plant conditions exist. These challenges include installation of jumpers to bypass valves, removal of RHR shutdown cooling system, extensive valve manipulation, system lineups, and procedural controls. Therefore, significant operational challenges and abnormal plant conditions would create a

more error-likely environment that could adversely impact the reactor safety.

Therefore, the licensee stated that the plant conditions discussed above would create hardship or unusual difficulty without a compensating increase in the level of quality and safety.

3.2 NRC Staff Evaluation

The NRC staff has evaluated this request for an alternative pursuant to 10 CFR 50.55a(z)(2). The NRC focused its review on whether compliance with the specified requirements of 10 CFR 50.55a(g), or portions thereof, would result in hardship or unusual difficulty, and if there is a compensating increase in the level of quality and safety despite the hardship.

3.2.1 Hardship

The NRC staff determined that requiring the licensee to comply with IWB-5221(a) to conduct the ASME Code pressure test of the Class 1 components identified in this request following repair/replacement activities in an unscheduled maintenance event or a forced outage event would result in hardship. The NRC staff determined the basis for hardship is as follows:

- High dose rates are expected in the drywell with the plant at 100 percent rated reactor power, which creates a hardship to personnel involved in conducting the associated VT-2 visual examination. In addition, the containment must be de-inerted for personnel entry. De-inerting the containment at 100 percent rated power could increase risk to safe operation of the plant, and impacts the plant TS requirements. Therefore, concerns from ALARA criteria and TS requirements, and potential operational risk that could adversely impact plant safety would constitute hardship.

- During plant start up at 5 percent of power, although containment access is permitted, extensive manipulations of the valves and controls are necessary to establish the required pressure of 1020 psig using nuclear heat. Manipulating the control systems during this plant condition would affect core reactivity and the normal range of operational limits and conditions. Changing the plant operational limits and conditions outside the normal range could cause significant risk to the plant safe operation and adversely impact the reactor safety. Therefore, significant operational challenges and risks to the reactor safety constitute hardship and unusual difficulty.

- Using decay heat and reactor recirculation pump heat before plant startup to pressurize the reactor to 1020 psig would require the RPV and the stream lines to be filled with coolant to provide an essentially water-solid condition. The licensee determined that extensive valves manipulations including installation of jumpers, system lineups, and procedural controls would be necessary to establish the required test pressure during outage conditions without withdrawal of the control rods. Maintaining the RCS pressure, temperature, heat-up, and cooldown rate within TS P-T limits without withdrawal of the control rods would create significant operational challenges, abnormal plant conditions, and

eventually an environment prone to error. Therefore, the NRC staff determines that potential operational risks that could adversely impact safe operation of the reactor constitute hardship.

As a result, the NRC staff concludes that concerns related to challenging the ALARA criteria, impacting TS requirements, approaching P-T limits, and abnormal plant conditions and potential operational risks that could adversely impact the reactor safety would constitute hardship.

3.2.2 Test Pressure

In evaluating the licensee's proposed alternative, the NRC staff assessed whether it appeared that the licensee used the highest test pressure to conduct pressure test of the Class 1 components in this request following repair/replacement activities in an unscheduled maintenance event or a forced outage event, and the manner in which the licensee adequately preformed the testing and the associated VT-2 visual examinations. The NRC staff determined that:

- The licensee will specifically conduct the test at a pressure of at least 888 psig, which is consistent with 87 percent of pressure required by IWB-5221(a).
- The test pressure of at least 888 psig will be attained by use of nuclear heat during the normal operational start-up sequence.
- Pressurization by nuclear heat to perform testing is less risky and safer for plant operation than the ASME Code pressure testing. In this approach, the reactor pressure and heat removal are controlled by the DEH system with steam being routed by the turbine bypass valves to the condenser, which is a normal pressure and temperature control mode for a BWR. The primary emergency core cooling systems are all operable.

The licensee will accomplish the proposed pressure testing without placing the plant in abnormal conditions, without exposing plant personnel to excessive radiation, and without significantly approaching the fracture toughness limits that could result in challenging the plant safe operation and personnel safety. Therefore, the NRC staff concludes that the licensee's proposed pressure test is adequate because the IWA-5240 required VT-2 visual examination with the proposed extended hold time will identify any evidence of leak following the repair/replacement activities.

3.2.3 Structural Integrity and Leak Tightness

In addition to the analysis described above, the NRC staff considered whether the licensee's proposed alternative provided reasonable assurance of structural integrity and leak tightness of the Class 1 components after repair/replacement activities.

The NRC staff notes that although the proposed test pressure is slightly lower than the ASME Code-required test pressure, it still allows for adequate pressurization of the components, and

as accompanied with the ASME Code-required VT-2 visual examination that will be performed after the proposed extended hold time, it allows for detection of potential leakage.

- As part of the pressure test, the licensee will perform the VT-2 visual examination of the components in accordance with the IWA-5240 requirements to identify any through-wall leak and/or mechanical joint leak.
- For non-insulated components, the VT-2 visual examination will be performed after attaining and holding test pressure for 1 hour to allow for leakage to accumulate at the potential leak location and be detected by the examination.
- For insulated components, the VT-2 visual examination will be performed after attaining and holding the test pressure for 8 hours to allow for leakage to accumulate at the potential leak location and be detected by the examination.

The NRC staff determines that with the longer hold times, the possibility of observing leakage will be increased, should a through-wall leak or a mechanical joint leak occur. The NRC staff also determines that the above hold times exceed the hold times specified in IWA-5213(a) and ASME Code Case N-795 and are, therefore, adequate.

Furthermore, the NRC staff determines that the proposed alternative (i.e., slightly reduced test pressure with increased hold time) provides reasonable assurance that any through-wall and/or joint leakage in the Class 1 components after repair/replacement activities will be identified by the VT-2 visual examination, and the licensee will take appropriate action.

Therefore, the NRC staff concludes that the proposed pressure test performed following the repair/replacement activities on the Class 1 components (excluding the RPV) accompanied by the associated VT-2 visual examinations performed after the proposed extended hold time is adequate to provide a reasonable assurance of structural integrity and leak tightness of the components. Complying with the requirement specified in IWB-5221(a) would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety.

The NRC staff authorizes the use of this alternative pressure test in the fourth 10-year ISI interval. The NRC staff's authorization of the proposed alternative in this request neither permits the use of this alternative pressure test to satisfy the requirements of Table IWB-2500, Examination Category B-P, nor to satisfy pressure test requirements of the RPV. Moreover, the authorization of the proposed alternative does not imply or infer the NRC approval of ASME Code Case N-795.

4.0 CONCLUSION

As set forth above, the NRC staff determines that the proposed alternative provides reasonable assurance of structural integrity and leak tightness of the Class 1 components, and complying with the specified requirement would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety. Accordingly, the NRC staff concludes that the licensee has adequately addressed all of the regulatory requirements set forth in 10 CFR 50.55a(z)(2). Therefore, the NRC staff authorizes the use of the licensee's proposed

alternative at CGS for the fourth 10-year ISI interval, which started on December 13, 2015, and is scheduled to end on December 12, 2025.

The authorization of the proposed alternative in relief request 4ISI-02 does not imply or infer the NRC approval of ASME Code Case N-795. All other ASME Code, Section XI, requirements for which relief was not specifically requested and approved herein by the NRC staff remain applicable, including third-party review by the Authorized Nuclear Inservice Inspector.

Principal Contributor: A. Rezai, NRR/DE/EPNB

Date: October 31, 2016

If you have any questions regarding this matter, please contact the NRC project manager, John Klos, at (301) 415-5136 or via e-mail at John.Klos@nrc.gov.

Sincerely,

/RA/

Robert J. Pascarelli, Chief
Plant Licensing Branch IV-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-397

Enclosure:
Safety Evaluation

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