ENERGY NORTHWEST

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July 14, 2016
GO2-16-090

10 CFR 50.55a

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, DC 20555-0001

Subject: **COLUMBIA GENERATING STATION, DOCKET NO. 50-397 RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION FOR RELIEF REQUEST 4ISI-02**

References:
1) Letter GO2-15-157, dated December 10, 2015, AL Javorik (Energy Northwest) to NRC, "Fourth Ten-Year Interval Inservice Inspection (ISI) Program Request 4ISI-02"
2) Email, dated June 23, 2016, John Klos (NRC) to Lisa Williams (Energy Northwest) "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"

Dear Sir or Madam:

By Reference 1, Energy Northwest submitted relief request 4ISI-02 for approval.

Via Reference 2, the Nuclear Regulatory Commission (NRC) requested additional information related to Energy Northwest's submittal. Enclosure 1 provides the requested information.

This letter and its enclosure contain no regulatory commitments. If there are any questions or if additional information is needed, please contact Ms. L. L. Williams, Licensing Supervisor, at 509-377-8148.

Executed this 14th day of July, 2016.

Respectfully,

FOR A.L. JAVORIK

A. L. Javorik
Vice President, Engineering

Enclosure: As stated

cc: NRC RIV Regional Administrator
NRC NRR Project Manager
NRC Senior Resident Inspector/988C
CD Sonoda - BPA1399 (email)
WA Horin – Winston & Strawn (email)

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION (RAI)

NRC Request:

RAI 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).

Energy Northwest RAI 1 Response:

Energy Northwest confirms that the alternative pressure test contained in Columbia Generating Station (Columbia) relief request 4ISI-02 will not be used to satisfy the requirements of Table IWB-2500-1, Examination Category B-P.

NRC Request:

RAI 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.

Energy Northwest RAI 2 Response:

The scope of relief request 4ISI-02 includes both isolable and non-isolable repair/replacement activities during an operating cycle between refueling outages. Nuclear heat would be used to raise primary system pressure to at least 887.4 psig (87% of 1020 psig as specified in Code Case N-795) with a hold time of one hour for non-insulated components and eight hours for insulated components which exceeds the hold times specified in Code Case N-795. This relief request would only be used after the Table IWB-2500-1 examination is complete for the refueling outage until the following refueling outage.

It is intended primarily for non-welded mechanical repair/replacement activities on components such as main steam relief valves (MSRVs), control rod drives (CRDs), reactor recirculation (RRC) pump mechanical seals, and in-core instrumentation. It would also be utilized for isolable and non-isolable welded repair/replacement activities.

Mechanical joints will be subjected to relief request 4ISI-02 in the following manner. If the alternative requirements to American Society of Mechanical Engineers (ASME) Section XI IWA-4000 which are contained in IWA-4131, IWA-4132, or ASME Code Case N-508-4 are utilized, no ASME Section XI pressure test and VT-2 visual examination will be performed. Repair/replacement activities associated with mechanical connections larger than National Pipe Standard (NPS) 1, not associated with the referenced alternative requirements above, will receive pressure testing and VT-2 examination in accordance with 10 CFR 50.55a(b)(2)(xxvi).

NRC Request:

RAI 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.

Energy Northwest RAI 3 Response:

Method 1

The first method would implement the reactor pressure vessel leakage test conducted in accordance with Columbia procedure OSP-RPV-R801 to meet the requirements of Table IWB-2500-1, Category B-P. This procedure also includes the 10-year pressure test where the test boundary includes all ASME Code Class 1 components. The reactor pressure vessel (RPV) leakage test normally occurs at the end of a refueling outage when core decay heat has had time to decrease. Some spent fuel has been removed and some new fuel added which results in a relatively low decay heat load. Heat up is performed using non-nuclear means (decay heat and RRC pump heat). The CRD pumps or an outside supplied hydro pump are used to pressurize the Class 1 piping and components. Pressure balancing and level control are provided by the CRD system and the blowdown mode of the reactor water clean-up (RWCU) system along with main steam drain orifice bypass valve MS-V-21 to the hotwell if the condenser is available. If the condenser is not available pressure balancing will be provided by the CRD system and RWCU blowdown to the radioactive equipment drains tank will be used for RPV level and pressure control The RPV is filled with coolant and the steam lines are flooded to the outboard main steam isolation valve (MSIV) to provide an essentially water-solid condition.

Residual heat removal (RHR) shutdown cooling is removed from service and isolated prior to reaching 48 psig to ensure operation of the RHR system does not exceed the analyzed value for the shutdown cooling mode of RHR. Thus the remaining system available for decay heat removal is the RWCU system via the non-regenerative heat exchanger. Other methods of slowing the heatup rate include increasing CRD flow and balancing with an increased reject flow to the main condenser, switching the CRD suction back to the condensate storage tank to inject colder water and increasing CRD flow (balanced with the reject flow to maintain a stable pressure), and reducing RRC pump speed to reduce pump heat input.

Use of this method requires extensive valve manipulations, system lineups, and procedural controls in order to heat up and pressurize the primary system to establish the necessary test pressure, during outage conditions, without the withdrawal of control rods while maintaining reactor coolant system (RCS) pressure, temperature, heatup rate and cooldown rate within Technical Specification limits. After completion of the testing and subsequent recovery from the test procedure, normal plant startup occurs.

Electrical jumpers are installed to bypass the MSIV closure on low condenser vacuum, to ensure the inboard MSIVs remain open throughout the test. Electrical jumpers are also installed on the Reactor Protection System (RPS) MSIV scram interlock and RPS input for RPV Steam Dome Pressure – High trip. These jumpers prevent an inadvertent scram signal induced by test conditions which would result in RPS actuation resulting in reportable condition. This test is performed in Mode 4 with the Reactor Mode Switch in shutdown. No other shutdown interlocks are bypassed.

Mechanical hose jumpers are installed if performing the 10-year pressure test to ensure the test pressure reaches the entire class 1 boundary by jumpering around system check valves. These include mechanical hose jumpers in the RHR system, the Reactor Feedwater (RFW) system, the low pressure core spray (LPCS) line, and the high pressure core spray (HPCS) line. Only some of these mechanical jumper installations may be required depending on where the component which requires testing is located. Additionally, the solenoid valves on the MSRVs are de-energized to protect against inadvertent lift of an MSRV during testing.

Using Method 1 during a short duration maintenance or forced outage, the higher decay heat creates a significant challenge to the operations staff while performing pressurization for the test. To support the test pressurization, RHR shutdown cooling would be required to be removed from service prior to reaching 48 psig as stated above leaving only the blowdown mode of RWCU system for decay heat removal. Isolating RHR shutdown cooling under high decay heat loads requires abnormal plant conditions and alignments and is accompanied by inherent risk. In addition, the hurried time frames, which result from high heat up rates, may create a more error-likely environment. For example, the ratio of decay heat input versus heat removal capacity provided by RWCU following one day and three days after shutdown would be approximately 5:1 and 3.6:1 respectively. The decay heat generated would surpass the capacity of RWCU and therefore, would delay performance of the IWB-5221(a) pressure test until a time when decay heat levels are lower. This could result in excessive delays of plant startup.

Method 1 could present other operational challenges as well. A failure of a component, such as the flow control valve or flow controller, would cause the interruption of RWCU blowdown flow and would cause the RPV pressure to increase. The RPV pressure would increase until operator action would require the operating CRD pump to be tripped. Due to the amount of decay heat being generated and the limited RWCU system heat removal capacity, the RPV may continue to pressurize and may require further operator action to depressurize the RPV. Operator actions may include one or more of the following: reestablish RWCU blowdown flow if the failure mechanism was no longer present, open the main steam line drain valves, head vent line, or reestablish the solenoid valves for the MSRVs and open an MSRV, or exit the surveillance and establish RHR shutdown cooling or alternate shutdown cooling.

Method 2

The second method would perform the pressure test and VT-2 examination during normal startup procedures. During normal startup with normal power ascension, nominal operating pressure of 1020 psig is reached at a reactor power level of approximately 100% rated thermal power (RTP). If access to containment were permitted at this power level, personnel would be exposed to excessive radiation levels, including significant exposure to neutron radiation levels, which is contrary to station as low as reasonably achievable (ALARA) practices. However, establishing the ASME Code required test condition at a more moderate power level such as during startup at approximately 5% to 10% RTP, and in the manner to address radiation concerns would require a significant change to the steam pressure control system.

Pressure control is accomplished by controlling main steam pressure immediately upstream of the main turbine throttle and governor valves through modulation of the turbine-governor or steam bypass valves. Command signals to these valves are generated by the digital electro-hydraulic (DEH) control system. For normal operation, the turbine governor valves regulate steam pressure. The plant's ability to change turbine generator output is enabled by adjusting reactor power level. In response to the resulting steam production changes, the DEH control system adjusts the turbine governor valves to accept the steam output change, thereby regulating steam pressure and changing turbine generator power output.

During performance of plant startup procedures, the DEH system is used to raise reactor pressure during power ascension up to normal operating pressure of 1020 psig at 100% RTP. The primary function is to regulate main steam pressure as sensed near the inlet of the high pressure turbine. The reactor pressure control at nominal 1020 psig is achieved at higher reactor power levels as a function of the pressure control system and the induced differential pressure across the main steam isolation valves and main steam piping.

While it is potentially technically feasible to manipulate these controls to establish the nominal system pressure of approximately 1020 psig at lower power levels, there is no nuclear analysis that supports this mode of operation, and doing so will affect core reactivity. Attempting to establish nominal operating pressure of 1020 psig at a lower power such as between 5% – 10% RTP is not a specifically analyzed plant operating condition. Reload licensing analysis for the fuel does not require consideration of plant conditions below 25% of rated with normal plant configurations, and hence would certainly not cover abnormal plant configurations as proposed by this possible scenario. Columbia has not previously operated the DEH system in this manner. Changing the setpoints outside the normal range of operation for the low power level of approximately 5% to 10% RTP, poses operational challenges. The lack of experience and predictability of setting the pressure regulators outside the normal range of operation challenges operations with the potential risk of adversely impacting reactor safety.

Method 3

The third method would be a modified Method 1 test which maintains RPV level at its normal operational level and uses decay heat to produce sufficient pressure to conduct the test at operating pressure. The MSIVs are closed as in Method 1 so the turbine valves would not be available for pressure control. Leaving the MSIVs open would require closing the turbine governor valves and controlling pressure using the turbine bypass valves using the DEH system. The same operational challenges would apply as identified in Method 2 above for operating the DEH system in this manner. The same operational challenges identified in Method 1 apply to Method 3. A short duration maintenance or forced outage with higher decay heat would create a significant challenge to the operations staff while performing pressurization for the test. To support the test pressurization, RHR shutdown cooling would be required to be removed from service prior to reaching 48 psig as stated above leaving only the blowdown mode of RWCU system for decay heat removal. Pressurization for normal system leakage examination would be provided by decay heat and the RRC pumps, with pressure balancing performed by the CRD system and blowdown mode of the RWCU system. Isolating RHR shutdown cooling under high decay heat loads requires abnormal plant conditions and alignments and is accompanied by inherent risk. In addition, the hurried time frames, which result from high heat up rates, may create a more error-likely environment. For example, the ratio of decay heat input versus heat removal capacity provided by RWCU following one day and three days after shutdown would be approximately 5:1 and 3.6:1 respectively. The decay heat generated would surpass the capacity of RWCU and therefore, would delay performance of the IWB-5221(a) pressure test which could cause excessive delay of plant startup.

Like Method 1, Method 3 could present other operational challenges as well. A failure of a component, such as the flow control valve or flow controller, would cause the interruption of RWCU blowdown flow and would cause the RPV pressure to increase. The RPV pressure would increase until operator action would require the operating CRD pump to be tripped. Due to the amount of decay heat being generated and the RWCU system heat removal capacity, the RPV may continue to pressurize and may require further operator action to depressurize the RPV. Operator actions may include one or more of the following: reestablish RWCU blowdown flow if the failure mechanism was no longer present, open the main steam line drain valves, head vent line, or reestablish the solenoid valves for the MSRVs and open an MSRV, or exit the surveillance and establish RHR shutdown cooling or alternate shutdown cooling.

Like Method 1, electrical jumpers are installed to bypass the MSIV isolation on low condenser vacuum, RPS MSIV scram interlock and RPS input for RPV Steam Dome Pressure – High trip.

Mechanical hose jumpers are installed as if performing the 10-year pressure test to ensure the test pressure reaches the entire class 1 boundary to jumper around system check valves. These include mechanical hose jumpers in the RHR system, the RFW system, the LPCS line, and the HPCS line. Only some of these mechanical jumper installations may be required depending on where the component which requires testing is located. Additionally, the solenoid valves on the MSRVs are de-energized to protect against inadvertent lift of an MSRV during testing.

In summary, each of the three methods discussed above that comply with IWB-5221(a) requirements for testing at 100% normal operating pressure present a hardship or unusual difficulty to Columbia without a compensating increase in quality and safety.

NRC Request:

RAI 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.

Energy Northwest RAI 4 Response:

The method for attaining and holding the 87% pressure requirement of the ASME Code Case N-795 using nuclear heat would be by normal startup sequences and procedures by raising RPV pressure to 888 psig corresponding to approximately 5% to 10% RTP. Power ascension is held at this level to meet the hold times and to permit access into containment by the VT-2 examination personnel.

To perform the test at approximately 5% to 10% RTP, RPV level is controlled by the feedwater level control system with reject flow to the RWCU. Pressure would be controlled by the DEH 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. No shutdown interlocks would be affected by this method. The plant would be in Mode 2 with the reactor mode switch in Startup/Hot Standby. The plant would be in a configuration with normal pressure, inventory and reactivity control.

Method 1 or 3 described in the response to RAI 3 above would be used for attaining and holding the 87% pressure requirement of the ASME Code Case N-795 without using nuclear heat. If nuclear heat is not used, then the primary source for increasing pressure would be decay heat. The DEH system is not available during shutdown since the turbine is isolated. The following table provides a summary as requested.

	Nuclear Heat Option	**Without Nuclear heat Option (Method 1 shown)**
Available methods and systems for heat removal (including heat removal capacity for each available system)	Heat is removed by directing steam to the main condenser hotwell. Approximately 7.7×10^9 BTU/hr	RHR shutdown cooling (Isolated at 48 psig): 47.6×10^6 BTU/hr (no tubes plugged) RWCU non-regenerative heat exchanger: 15.09×10^6 BTU/hr (5-U-Tubes plugged)
Available methods and systems for pressure control	DEH system with steam via the turbine bypass valves to the condenser The turbine bypass control is set to AUTO when RPV pressure is approximately 30 psig.	CRD system and letdown to the RWCU system along with main steam drain orifice bypass valve if main condenser is available
Available methods and systems for inventory control	Feedwater system with RWCU reject flow	CRD system injection with letdown to the RWCU system
Available methods and systems for reactivity control	Control Rods	All rods are in except during scram time testing since this method would be performed with the reactor shutdown.
An indication of the mode and plant operating state	Mode 2, Reactor Mode Switch in Startup/Hot Standby	Mode 4
Any changes to normal interlocks	No interlock are affected by this method	See Method 1 discussion under RAI 3 response.

	Nuclear Heat Option	Without Nuclear heat Option (Method 1 shown)
An indication of whether the heat removal, pressure, inventory, and reactivity control would be sufficient for the configuration of the plant	Normal plant startup conditions with normal plant configuration for heat removal, pressure, inventory, and reactivity control.	Heat removal and pressure control would be limited for this plant configuration. Heat removal is not sufficient until decay heat levels have dropped.

NRC Request:

RAI 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.

Energy Northwest RAI 5 Response:

A review was conducted of repair/replacement activities performed during the past five operating cycles. There has been one repair at Columbia for which Relief Requests 4ISI-02 or 3ISI-12 could have been used if they were available at the time of the repair: CRD 14-31 was replaced due to a malfunction. The CRD is attached via an approximately 6” diameter mechanical flange connection which is non-isolable. An initial VT-2 examination at an RPV pressure of 935 psig was performed during the forced outage when the CRD mechanism was replaced with no leakage observed. However, since this initial examination was performed at 935 psig, it did not meet the ASME Section XI IWB-5210 requirement to perform the test at a pressure corresponding to 100% RPT. This was documented in the Energy Northwest corrective action program. The final VT-2 examination which met the ASME code required pressure corresponding to 100% RPT was performed during the Table IWB-2500-1, examination category B-P, required system leakage test at the end of the next refueling outage. The VT-2 visual examination was performed successfully when the RPV pressure was greater than or equal to 1020 psig with no leakage observed.