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

March 1, 2013

Site Vice President
Entergy Nuclear Operations, Inc.
Vermont Yankee Nuclear Power Station
P.O. Box 250
Governor Hunt Road
Vernon, VT 05354

SUBJECT: VERMONT YANKEE NUCLEAR POWER STATION – RELIEF REQUEST ISI-PT-02: FOURTH 10-YEAR INSERVICE INSPECTION INTERVAL (TAC NO. MF0423)

Dear Sir or Madam:

By letter to the U.S. Nuclear Regulatory Commission (NRC) dated December 21, 2012 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML12362A013) as supplemented by letter dated February 5, 2013 (ADAMS Accession No. ML13038A008), Entergy Nuclear Operations, Inc. requested relief from the requirements of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code), Section XI, IWC-5221, for the fourth 10-year inservice inspection (ISI) interval at the Vermont Yankee Nuclear Power Station.

Specifically, pursuant to Title 10 of the *Code of Federal Regulations* (10 CFR) 50.55a(a)(3)(ii), the licensee requested to use the alternative in Relief Request ISI-PT-02 on the basis that complying with the specified ASME Code requirement would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety.

The details of the NRC staff review are included in the enclosed safety evaluation. The NRC staff concludes that the licensee has adequately addressed all of the regulatory requirements set forth in 10 CFR 50.55a(a)(3)(ii) and, therefore, is in compliance with the ASME Code requirements.

Therefore, the licensee's proposed alternative is authorized in accordance with 10 CFR 50.55a(a)(3)(ii) at VY.

Sincerely,

Sean C. Meighan, Acting Branch Chief
Plant Licensing Branch I-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-271

Enclosure: Safety Evaluation

cc w/enclosure: Distribution via ListServ

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

RELIEF REQUEST ISI-PT-02: FOURTH 10-YEAR INSERVICE INSPECTION INTERVAL

ALTERNATIVE TO SYSTEM LEAKAGE TEST FOR THE

REACTOR PRESSURE VESSEL HEAD FLANGE LEAK-OFF LINES

ENTERGY NUCLEAR OPERATIONS, INC.

VERMONT YANKEE NUCLEAR POWER STATION

DOCKET NUMBER 50-271

1.0 INTRODUCTION

By letter dated December 21, 2012 (Agencywide Documents and Access Management System (ADAMS) Accession No. ML12362A013) as supplemented by letter dated February 5, 2013 (ADAMS Accession No. ML13038A008), Entergy Nuclear Operations, Inc. (Entergy or the licensee) requested relief from the requirements of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code), Section XI, IWC-5221, for the Vermont Yankee Nuclear Power Station.

Specifically, pursuant to Title 10 of the *Code of Federal Regulations* (10 CFR) 50.55a(a)(3)(ii), the licensee requested to use the alternative in Relief Request ISI-PT-02 on the basis that complying with the specified ASME Code requirement would result in hardship or unusual difficulty. Relief Request ISI-PT-02 is applicable to the system leakage test of the reactor pressure vessel (RPV) head flange leak-off lines at the Vermont Yankee Nuclear Power Station for the fourth 10-year inservice inspection (ISI) interval.

2.0 REGULATORY EVALUATION

Pursuant to 10 CFR 50.55a(g)(4), ASME Code Class 1, 2, and 3 components (including supports) must 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 (ISI) 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 interval and subsequent 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 interval, subject to the conditions listed therein.

Enclosure

Section 50.55a(a)(3) of 10 CFR, states, in part, that alternatives to the requirements of paragraph (g) of 10 CFR 50.55a may be authorized by the NRC if the licensee demonstrates that: (i) the proposed alternative provides 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.

Based on the above, and subject to the following technical evaluation, the NRC staff finds that regulatory authority exists for the licensee to request and the NRC staff to authorize the alternative requested by the licensee.

3.0 TECHNICAL EVALUATION

3.1 RELIEF REQUEST ISI-PT-02

3.1.1 ASME Code Components Affected

The affected piping system is the reactor pressure vessel head flange leak-off lines originating from reactor vessel nozzles N-13 and N-14. The affected components include valve FCV-2-20 (FCV-20), valve FCV-2-21 (FCV-21), orifice RO-2-23, valve V2-22, excess flow check valve SL-2-23, pressure instrument PI-2-101 (PI-101), pressure switch PS-2-102 (PS-102), and associated piping and fittings. The affected components are classified as ASME Code Class 2, Examination Category, C-H, Item Number, C7.10.

By letter dated February 5, 2013, the licensee stated that:

> The material specification for the seamless stainless steel pipe is ASTM A-376 or A-312, grade TP304 or TP316. The weld filler metal complies with ASME SA-371 or ASTM A371.

The licensee further stated that:

> The design pressure of the 1-inch and ½-inch diameter piping is 1250 pounds per square inch gauge (psig). The wall thickness for both diameters is schedule 160.

3.1.2 Applicable Code Edition and Addenda

The applicable code edition and addenda is ASME Boiler and Pressure Vessel Code, Section XI, 2001 Edition through 2003 Addenda.

3.1.3 Applicable Code Requirement (as stated by licensee)

> [The ASME Code, Section XI] IWC-5221: The system leakage test shall be conducted at the system pressure obtained while the system, or portion of the system, is in service performing its normal operating function or at the system pressure developed during a test conducted to verify system operability (e.g. to

demonstrate system safety function or satisfy technical specification surveillance requirements).

3.1.4 Reason for Request (as stated by the licensee)

The ASME Boiler and Pressure Vessel Code requirement for the ASME Code Class 2 reactor pressure vessel (RPV) head flange leak-off lines is to perform a system leakage test at a pressure corresponding to 100 percent rated power. This is approximately 1005 psig. However, the configuration of the leak-off lines makes testing difficult and presents a personnel and equipment hazard.

The licensee further stated that:

The RPV head flange leak-off line originating from reactor vessel nozzle N13 (N13 leak-off line) is separated from the reactor coolant pressure boundary by one passive membrane, which is a metallic seal ring located on the inner vessel flange. A second seal ring is located on the outside of the tap in the vessel flange. The leak-off line originating from reactor vessel nozzle N14 (N14 leak-off line) has a tap located outside of the second seal ring. For the N13 leak-off line, failure of the inner seal ring is the only condition under which this line is pressurized. The N14 leak-off line requires both seal rings to fail. Therefore, these lines are not expected to be pressurized during normal reactor operation.

The licensee further stated that:

The configuration of this piping precludes system leakage testing while the RPV head is removed because the configuration of the vessel taps, coupled with the high test pressure, prevents the taps from having test equipment installed. The N13 leak-off line tap has a diameter of 3/16 inches and the N14 leak-off line tap has a diameter of ½ inches. Both taps are smooth walled making the effectiveness of a temporary test connection seal at high pressure limited. Failure of the test connection presents a personnel safety hazard and may result in foreign material entering the reactor vessel.

The licensee explained that:

The configuration also precludes pressurizing the lines with the RPV head installed. The head contains two grooves that hold the seal rings. The seal rings are held in place by a series of retainer clips that are housed in recessed cavities in the flange face. If a system leakage test was performed via the N13 leak-off line, the inner seal ring would be pressurized in a direction opposite its design function. This test pressure would result in a net inward force on the inner seal ring that would tend to push it into the recessed cavity that houses the retainer clips. The thin seal ring material would likely be damaged by the inward force. The same situation exists for the outer seal ring when testing via the N14 leak-off line.

According to the licensee:

> Not installing the seal rings in order to perform a system leakage test requires the additional time and radiation exposure associated with the subsequent removal and reinstallation of the RPV head to install the seal rings prior to startup. Purposely failing the seal rings would require purchasing new seal rings in addition to the subsequent removal and reinstallation of the RPV head.

The licensee concluded that:

> Considering the above information, compliance with IWC-5221 will result in undue hardship and unusual difficulty without a compensating increase in the level of quality or safety.

3.1.5 Proposed Alternative and Basis for Use (as stated by the licensee)

> In lieu of the requirements of [The ASME Code, Section XI], IWC-5221, a VT-2 visual examination will be performed during the next refueling outage in March 2013 on the accessible portions of the affected components. This is the last refueling outage in the fourth inservice inspection (ISI) program interval. The examination will be conducted with the affected components subject to static pressure head with the RPV head removed and the refueling cavity filled to its normal refueling water level for at least four (4) hours. The static head developed with the leak-off lines filled with water will allow for detection of pressure boundary failures.

In the February 5, 2013, letter, the licensee stated that:

> When at the normal refueling water level (elevation approximately 343 feet, 6 ¾ inches), the static head pressure at the RPV head flange (elevation 321 feet, 6 inches) will be approximately 9.5 psig.

The licensee stated that:

> This proposed alternative is identical to that presented in ASME Code Case N-805. However, this Code Case has not yet been approved by the NRC and is not yet identified in Regulatory Guide 1.147, Revision 16, "In-service Inspection Code Case Acceptability, ASME Section XI, Division 1."

3.1.6 Duration of Proposed Alternative

The licensee requested this alternative for the remainder of the fourth ISI program interval, which ends on August 31, 2013.

3.2 NRC Staff Evaluation

As stated above, the affected components are classified as Examination Category C-H in accordance with the 2001 edition through 2003 Addenda of the ASME Code, Section XI, Table IWC-2500-1. Examination Category C-H requires a system leakage test be performed in accordance with IWC-5220 and accompanied with a VT-2 visual examination once every inspection period. The ASME Code, Section XI, IWC-5221 requires use of normal operation pressure (approximate 1005 psig) to perform the system leakage test. In lieu of using 1005 psig pressure, the proposed alternative will use the static head of the reactor cavity water of approximate 9.5 psig to perform the system leakage test for the leak-off lines. Given a test pressure of 9.5 psig, the NRC staff focuses its evaluation on how the licensee performs the associated visual examination and how the potential leakage can be detected to demonstrate the structural integrity and leak tightness of the leak-off lines.

3.2.1 Leak-off Line Configuration

The reactor vessel head flange leak-off lines have two taps N13 and N14 at the reactor vessel head flange as shown in the piping and instrumentation diagram in the December 21, 2012, submittal. For each of the N13 and N14 taps, a bore hole is drilled through the RPV head lower flange. The N13 tap is a 3/16 inch diameter hole drilled between the seal rings. The N14 tap is a ½ inch diameter hole drilled on the outside of the outer seal ring. The primary leak-off line is connected to the N13 tap and this line is equipped with flow control valves, pressure switch PS-102, pressure instrument PI-101 and orifices as shown in the submittal.

As discussed in the February 5, 2013, letter, for normal system alignment and configuration:

> Valve FCV-20 located in the drywell is normally open and valve. FCV-21 located in the drywell is normally closed. Manual isolation valve V-22 and excess flow check valve SL-23 located on reactor building elevation 280 feet are normally open. Pressure indicator PI-101 and pressure switch PS-102 are both located outside the control room on reactor building elevation 280 feet on Rack 25-5.

The licensee stated that:

> During normal plant operation, the line originating from N13 is not expected to pressurize from leakage through the seal rings. However, following a refueling outage when the line becomes filled, the line will pressurize due to heatup of the trapped water. This is an expected condition.

The secondary leak-off line is connected to the N14 tap and the N14 line is separate from the line originating from N13. In order for the N14 line to become pressurized, both the inner and outer seal rings need to fail. If the N14 line becomes pressurized, the inner seal has failed and would pressurize the N13 line. This line would then be monitored for leakage past the outer seal ring. Drywell entry would be required to install the required equipment and instrumentation.

The NRC staff finds that the licensee has clarified the configuration and system alignment of the leak-off lines and identified the significant components.

3.2.2 Scope of Examination (as stated by the licensee)

By letter dated February 5, 2013, the licensee identified:

> The provided markup of the piping and instrumentation diagram highlights the components [the portion of the leak-off lines] required to be examined in accordance with the ASME Code, Section XI, IWC-5000. These consist of the Class 2 piping originating from nozzle N13 up to valve FCV-21 and out to the pressure switch PS-102 and pressure indicator PI-101.
>
> Also required to be examined is all Class 2 piping originating from nozzle N14 out to the cap [as shown in the piping and instrumentation diagram].
>
> The non-class piping downstream of valve FCV-21 is not required to be examined in accordance with the ASME Code, Section XI, IWC-5000.

The licensee clarified that:

> It is unknown at this time which pipe segments inside the drywell are not accessible. The licensee stated that all segments of the instrumentation piping in the reactor building are accessible.
>
> As permitted by ASME Code, Section XI, IWA-5241(b), for any pipe segments that are inaccessible, a VT-2 examination of the surrounding area will be performed for evidence of leakage. This includes floor areas or equipment surfaces located underneath the pipe segment or other areas where leakage may be channeled.

The NRC staff finds that the licensee has appropriately identified the scope of required examination and that the licensee will perform VT-2 examination in accordance with the ASME Code, Section XI, IWA-5240 to look for leakage during the system leakage test of the leak-off line.

3.2.3 Proposed System Leakage Test (as stated by the licensee)

As discussed in the February 5, 2013, letter, for the proposed alternative, the licensee will perform:

> The VT-2 examination will be performed in accordance with Entergy procedure CEP-NDE-090 [during the system leakage test]. The examination will begin after the RPV head has been removed and the reactor cavity has been filled to its normal refueling level for at least four hours.
>
> A procedure demonstration using a character card or light meter will be performed to verify suitable lighting conditions. This demonstration will be

> performed at the examination location or at a simulated location for remote examination.
>
> For non-insulated sections of the leak-off line, the VT-2 examination will be conducted by examining the accessible external exposed surfaces for evidence of leakage. Evidence of leakage includes areas of general corrosion. If access is limited, the [licensee] Operator will improve access by use of items such as mirrors or ladders. Remote examination using items such as binoculars or scopes is permitted.
>
> For insulated sections of the leak-off line, the VT-2 examination will be conducted [by the licensee] without the removal of insulation by examining the accessible and exposed surfaces and joints of the insulation. Vertical surfaces of insulation will be examined [by the licensee] at the lowest elevation where leakage may be detectable. Horizontal surfaces of insulation will be examined [by the licensee] at each insulation joint.
>
> For the pipe sections that are inaccessible for direct VT-2 examination, an examination of the surrounding area will be performed for evidence of leakage [by the licensee]. This includes floor areas or equipment surfaces located underneath the pipe section or other areas where leakage may be channeled.

The NRC staff asked the licensee to demonstrate the structural integrity and leak tightness of the unexamined portion of the leak-off lines.

By letter dated February 5, 2013, the licensee responded that:

> There is no site-specific history of degradation identified on this line at the plant. However, stainless steel piping is known to be susceptible to stress corrosion cracking. There were four instances of industry experience identified documenting cracking on this line.

The NRC staff finds that the four instances of stainless steel cracking may or may not apply to the leak-off lines at Vermont Yankee, depending on various applied stresses and environment. Should the degradation in the four instances apply to the leak-off lines at Vermont Yankee and the leak-off lines do degrade in the future, the NRC staff finds that the periodic system leakage tests and visual examinations will provide sufficient monitoring of the structural integrity of the unexamined portion of the leak-off lines.

The NRC staff finds that the licensee's VT-2 examination is consistent with the requirements of the ASME Code, Section XI, IWA-5240.

The NRC staff asked how the operator distinguishes various leakage sources such as the normal leakage from the RPV passing through the seal rings, in-line leakage from the closed valves, bolt connections, or leakage from the degraded pipe wall.

By letter dated February 5, 2013, the licensee explained that:

> Leakage from the RPV through the seal rings is not considered normal and is not expected to occur. Operators are trained to identify leakage from plant components and their indications that may be present. All identified leaks are promptly investigated [by the Operator regardless of its origin, and have a condition report initiated] to determine their source. The source of the leakage may not always be readily identified by the Operator. There may be situations where insulation removal is necessary or scaffolding is required.

The licensee will perform corrective actions depending upon the leak location and severity. The NRC staff finds it is acceptable that the licensee has procedures to investigate various leakage sources.

<u>Leakage Detection Capability</u>

The NRC staff questioned how the licensee can detect the leakage in the leak-off lines during normal operation in case the proposed alternative was not able to detect through wall leakage during refueling outages.

In the February 5, 2013, letter, the licensee explained that:

> Pressure switch PS-102 is set at approximately 613 psig. Upon exceeding this setpoint, an alarm is actuated in the Control Room on Panel CRP 9-4.
>
> Upon actuation of the alarm in the Control Room, alarm response procedure, ARS 21002 will be entered. "....the first action is to have an Operator verify the condition by observing pressure indicator PI-101. [Pressure instrument PI-101 is located on reactor building elevation 280 feet on Rack 25-5 and is readily accessible to the Operator].

Following verification, Control Room Personnel will cycle solenoid-operated valves FCV-20 and FCV-21 in order to direct the pressurized water to the Drywell Equipment Sump.

The licensee further responded that:

> Continuing through the alarm response procedure, the Operator will continue to monitor the pressure indicator for a subsequent pressure rise. Control Room Personnel will monitor Drywell sump leakage for evidence of leakage.
>
> If pressure remains high, the valves will not be cycled again;[the licensee] plant management will then decide if plant operation can continue.

The NRC staff asked that if RPV head flange leakage causes pressurization of the leak-off lines and the lines depressurize after the alarm pressure level is tripped, describe the procedures to monitor the situation and corrective actions.

By letter dated February 5, 2013, the licensee responded

> ...even if the alarm immediately clears, there will still be indication in the Control Room that the alarm had actuated.
>
> The alarm response procedure does not provide exacting direction for this scenario which assumes the pressure indicator PI-101 indicates 0 psig. However, plant operations personnel are trained to consider all options that would provide those indications and alarms. It would be considered [by the Operator] that either the pressure switch had failed or a line rupture had occurred.
>
> A functional test of the pressure switch per procedure RP 4399 would likely occur also.

If through-wall leakage occurs in the leak-off lines, the licensee will assume either the inner seal ring or both seal rings have failed and the issue will be captured in the corrective action program. During normal plant operation with the drywell closed the licensee stated:

> If leakage occurred [in the section of the leak-off lines located] in the Drywell, the leak would manifest itself as an increase in unidentified drywell leakage (sump volume), Drywell temperature, and/or drywell radiation levels [that are indicated in the Control Room].

According to the licensee, if the leakage was large enough to affect those parameters, the leakage would be detected within one shift.

> If the leakage occurred in the [section of the line located in the] reactor building (outside the drywell), it would be identified as a steam plume, water dripping from equipment, an increase in the reactor building temperature, and/or radiation alarms [by the roving operator].
>
> Corrective actions would be based on the location and severity of the leak and may include a plant shutdown to perform a drywell entry.

The NRC staff determines that the licensee has sufficient leakage detection capability for the leak-off lines because (1) the alarm of the leakage detection system is available to the operators in the control room, and (2) the operator has procedures to identify and disposition the leakage. Therefore, even if, or in case, the proposed alternative was not able to identify a through wall flaw, the existing procedures and leakage detection capability will be able to identify the leakage during normal operation and the licensee will take appropriate corrective actions.

The NRC staff finds that the proposed alternative using low test pressure, although it may not be as effective as using the normal RCS operating pressure, will provide reasonable assurance of the structural integrity and leak tightness of the leak-off lines.

3.2.4 Hardship

In addition to the hardship discussed above, in the February 5, 2013, letter, the licensee stated that:

> It is estimated from historical data that approximately 5 [roentgen equivalent man] (rem) in additional dose would be received if the system leakage test was performed in accordance with the ASME Code, Section XI, IWC-5000. The licensee estimated this radiation dose assuming the test was performed with both the inner and outer seal rings removed requiring that the RPV head be subsequently removed to install the seal rings for normal plant operation. The RPV head would then be reinstalled for the second time. The licensee estimated that the radiation dose received using the proposed alternative will be less than 100 mrem.

In summary, the NRC staff finds that the licensee has provided sufficient basis to demonstrate that performing the system leakage test in accordance with the ASME Code, Section XI, IWC-5220 would result in a hardship and unusual difficulty without a compensating increase in quality and safety. The NRC staff further finds that the proposed alternative will provide reasonable assurance that the structural integrity and leak tightness of the reactor vessel head flange leak-off lines will be maintained for the remainder of the fourth 10-year ISI interval.

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 reactor vessel flange leak-off lines. The NRC staff finds that complying with the specified ASME Code 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(a)(3)(ii) and is in compliance with the requirements of the ASME Code, Section XI, for which relief was not requested. Therefore, the NRC authorizes the use of Relief Request ISI-PT-02 at the Vermont Yankee Nuclear Power Station for the remainder of the fourth 10-year ISI interval which ends on August 31, 2013.

All other ASME Code, Section XI requirements for which relief has not been specifically requested and approved in this relief request remain applicable, including third party review by the Authorized Nuclear Inservice Inspector.

Principal Contributor: John Tsao

Date: March 1, 2013

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March 1, 2013

Site Vice President
Entergy Nuclear Operations, Inc.
Vermont Yankee Nuclear Power Station
P.O. Box 250
Governor Hunt Road
Vernon, VT 05354

SUBJECT: VERMONT YANKEE NUCLEAR POWER STATION – RELIEF REQUEST ISI-PT-02: FOURTH 10-YEAR INSERVICE INSPECTION INTERVAL (TAC NO. MF0423)

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Therefore, the licensee's proposed alternative is authorized in accordance with 10 CFR 50.55a(a)(3)(ii) at VY.

Sincerely,
/ra/
Sean C. Meighan, Acting Branch Chief
Plant Licensing Branch I-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-271

Enclosure: Safety Evaluation

cc w/enclosure: Distribution via ListServ

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