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

February 18, 2010

Mr. Paul A. Harden
Site Vice President
FirstEnergy Nuclear Operating Company
Beaver Valley Power Station
Mail Stop A-BV-SEB1
P.O. Box 4, Route 168
Shippingport, PA 15077

SUBJECT: BEAVER VALLEY POWER STATION, UNIT NO. 1 - RELIEF REQUEST NUMBER 1-TYP-4-B3.120-1 REGARDING INSPECTION OF PRESSURIZER SURGE LINE NOZZLE (TAC NO. ME1108)

Dear Mr. Harden:

By letter dated April 14, 2009, as supplemented by letter dated October 19, 2009, FirstEnergy Nuclear Operating Company (licensee) submitted a request for authorization of a proposed alternative to American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code), Section XI, Table IWB-2500-1 for volumetric examination or enhanced visual examination requirements of pressurizer surge line nozzle at Beaver Valley Power Station, Unit No. 1 (BVPS-1) for the fourth 10-year inservice inspection (ISI) interval. The licensee also requested authorization of a proposed alternative to ASME Code, Section XI, Table IWC-2500-1 for surface examination requirements of accessible portions of the recirculation spray pump casting welds at Beaver Valley Power Station, Unit No. 2, in Relief Request No. 2-TYP-3-C6.10-1. However, by letter dated October 19, 2009, the licensee withdrew Relief Request No. 2-TYP-3-C6.10-1.

The Nuclear Regulatory Commission (NRC) staff has concluded that compliance with the specified ASME Code would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety, and that the proposed visual examination, in combination with walkdowns during each refueling outage, provides reasonable assurance of the structural integrity of the pressurizer surge line nozzle. Therefore, pursuant to Title 10 of the *Code of Federal* Regulations 50.55a(a)(3)(ii), the NRC staff authorizes the proposed alternative for BVPS-1 for the fourth 10-year ISI interval.

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

If you have any questions, please contact the Beaver Valley Project Manager, Nadiyah Morgan, at (301) 415-1016.

Sincerely,

for,

Nancy L. Salgado, Chief
Plant Licensing Branch I-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-334

Enclosure:
As stated

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

REGARDING THE FOURTH 10-YEAR INSERVICE INSPECTION PLAN INTERVAL

RELIEF REQUEST NUMBER 1-TYP-4-B3.120-1

FIRSTENERGY NUCLEAR OPERATING COMPANY

FIRSTENERGY NUCLEAR GENERATION CORP.

BEAVER VALLEY POWER STATION, UNIT NO. 1

DOCKET NO. 50-334

1.0 INTRODUCTION

By letter dated April 14, 2009 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML091070244), as supplemented by letter dated October 19, 2009 (ADAMS Accession No. ML092950338), FirstEnergy Nuclear Operating Company (licensee) submitted a request for authorization of a proposed alternative to American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code), Section XI, Table IWB-2500-1 for volumetric examination or enhanced visual examination requirements of pressurizer surge line nozzle at Beaver Valley Power Station, Unit No. 1 (BVPS-1) for the fourth 10-year inservice inspection (ISI) interval. The licensee also requested authorization of a proposed alternative to ASME Code, Section XI, Table IWC-2500-1 for surface examination requirements of accessible portions of the recirculation spray pump casting welds at Beaver Valley Power Station, Unit No. 2, in Relief Request No. 2-TYP-3-C6.10-1. However, by letter dated October 19, 2009, the licensee withdrew Relief Request No. 2-TYP-3-C6.10-1.

2.0 REGULATORY EVALUATION

Pursuant to Section 50.55a(g)(4) of Part 50 to Title 10 of the *Code of Federal Regulations* (10 CFR), ASME Code Class 1, 2, and 3 components (including supports) will meet the requirements, except the design and access provisions and the pre-service examination requirements, set forth in the ASME Code, Section XI, Rules for 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) twelve months prior to the start of the 120-month interval, subject to the limitations and modifications listed therein.

10 CFR 50.55a(b)(2)(xxi) *Table IWB-2500-1 examination requirements,* applies and requires that "(A) The provisions of Table IWB-2500-1, Examination Category B-D, Full Penetration Welded Nozzles in Vessels, Items B3.40 and B3.60 (Inspection Program A) and Items B3.120 and B3.140 (Inspection Program B) of the 1998 Edition must be applied when using the 1999 Addenda through the latest edition and addenda incorporated by reference in paragraph (b)(2) of this section. A visual examination with magnification that has a resolution sensitivity to detect a 1-mil width wire or crack, utilizing the allowable flaw length criteria in Table IWB-3512-1, 1997 Addenda through the latest edition and addenda incorporated by reference in paragraph (b)(2) of this section, with a limiting assumption on the flaw aspect ratio (i.e., a/l = 0.5), may be performed instead of an ultrasonic examination."

Pursuant to 10 CFR 50.55a(a)(3), alternatives to requirements may be authorized by the Nuclear Regulatory Commission (NRC) if the licensee demonstrates that: (i) the proposed alternatives 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.

The code of record for BVPS-1 for the fourth ISI interval is the 2001 Edition through the 2003 Addenda of the ASME Code, Section XI.

3.0 TECHNICAL EVALUATION

3.1 System/Component Affected

Pressurizer Surge Nozzle Inner Radius Section RC-TK-1-RADIUS-6 at BVPS-1

3.2 Applicable Code Edition and Addenda

The applicable Code is the ASME Code, Section XI, 2001 Edition through the 2003 Addenda and the 1998 Edition with the limitations and modifications of 10 CFR 50.55a(b)(2)(xxi).

3.3 Applicable Code Requirements

Table IWB-2500-1, Examination Category B-D, Inspection Program B, Item No. B3.120 requires volumetric examination of pressurizer nozzle inside radius section. 10 CFR 50.55a(b)(2)(xxi) permits enhanced visual examination of the exterior surface instead of ultrasonic examination as stated below.

(xxi) *Table IWB-2500-1 examination requirements.* "(A) The provisions of Table IWB-2500-1, Examination Category B-D, Full Penetration Welded Nozzles in Vessels, Items B3.40 and B3.60 (Inspection Program A) and Items B3.120 and B3.140 (Inspection Program B) of the 1998 Edition must be applied when using the 1999 Addenda through the latest edition and addenda incorporated by reference in paragraph (b)(2) of this section. A visual examination with magnification that has a resolution sensitivity to detect a 1-mil width wire or crack, utilizing the allowable flaw length criteria in Table IWB-3512-1, 1997 Addenda through the latest edition and addenda incorporated by reference in paragraph (b)(2) of this section, with a limiting assumption on the flaw aspect ratio (i.e., a/l = 0.5), may be performed instead of an ultrasonic examination."

3.4 Licensee's Basis for Request

In accordance with 10 CFR 50.55a(a)(3)(ii), the licensee proposed an alternative on the basis that compliance with the ASME Code requirement would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety. The difficulty associated with access to either the exterior surface for the volumetric examination or to the interior surface for the enhanced visual examination is described below. The BVPS-1 pressurizer surge nozzle is integrally cast within the SA216 Grade WCC lower head. To perform the volumetric examination of the surge nozzle inner radius section, the outside surface of the lower head must be accessible. This surface is made accessible by removing the insulation surrounding the surge nozzle. The design of this insulation requires disconnecting the 78 heater cables from the immersion heaters prior to removing the insulation. Each cable consists of two wires, each mechanically connected to the heater. Care must be taken while disconnecting and reconnecting to ensure the ceramic terminal blocks connecting the wires to the heater pins are not damaged. If damaged, an unbrazing/brazing evolution would be required to replace the blocks. The dose estimate for this examination assumes that no ceramic terminal blocks would require replacement. The licensee stated that another concern is the presence of asbestos in the cable jackets. Special monitoring and material control would be necessary due to the presence of friable asbestos. Though airborne radioactive contamination is not typically a concern in this area, respirators would be required to address the potential asbestos exposure. Additional cover-alls, suitable for asbestos work, would be required over the anti-contamination clothing, causing a heat stress concern.

The licensee stated that, "the dose estimate for cable disconnection/reconnection, insulation removal/reinstallation, surface preparation, and ultrasonic examination is 17.7 Rem. The dose estimate is based on a radiological survey conducted with the insulation installed. Once the insulation is removed the dose rates documented in the survey would likely increase. Shielding at this location is not practical since the radiation source is the component surface to be examined."

The licensee also stated that dose estimate "is based on a radiological survey obtained on February 13, 2006, during the seventeenth refueling outage (1R17). The hours estimated to perform the activities involved in disconnecting and reconnecting the cables are based on similar efforts performed during the eighth refueling outage (1R08) when two cables were reworked due to potential short circuits. Comprehensive dry-run exercises were performed on a mock-up in preparation for the 1R08 efforts. Therefore, the estimated times used in the dose estimate are believed to be reasonably accurate. With the nozzle insulation removed from this area, the complete code required examination could be performed. Special equipment was designed to perform this specific examination. The other five top-head pressurizer inner radius sections have been successfully examined. No recordable indications were noted on these examinations. Previous examinations of the adjacent safe-end to nozzle weld have been completed with satisfactory results. The next examination on this safe end-to-nozzle weld will be performed during the nineteenth refueling outage (1R19) in the spring of 2009. BVPS-1 and 2 have over 50 years of combined operation, with no leakage in the surge nozzle area. The BVPS- 2 surge nozzle inner radius section and the surge nozzle-to-vessel weld were ultrasonically examined during the previous interval and no recordable indications were found. The BVPS-1 surge nozzle inner radius section has not been examined."

The licensee evaluated a remote visual examination from the inside of the pressurizer as an alternative to the ultrasonic examination. A screen located at the surge line nozzle and baffle plates in the lower section of the pressurizer restricts access to the area of interest. The surge nozzle contains a thermal sleeve that would significantly limit the examination, if access through the screen were possible. The distance from the manway to the surge nozzle area is approximately 40 feet, making positioning adjustments of the remote camera difficult. Because of these limitations, the licensee stated that a remote visual examination from the inside of the pressurizer is not considered a viable alternative.

The licensee stated that several methods are available to detect leakage from this and other areas if a throughwall leak were to occur.

a. Control room operators perform Operation Surveillance Test (OST) 1OST-6.2, "Reactor Coolant System Water Inventory Balance," every 72 hours with the plant operating at steady state conditions. The operators would discover the leakage by the conduct of this test.

b. Containment airborne radiation monitors continuously sample the containment atmosphere and alarm in the control room. This detection method is dependent on the size of the leak, reactor coolant activity, and containment background activity.

c. Leakage would cause an increase in containment pressure, temperature and humidity, which are indicated in the control room. Additionally, high containment pressure produces an alarm in the control room.

d. Substantial leakage would collect in the containment sump. The sump level is indicated and alarmed in the control room.

The licensee contends that there are no credible failure mechanisms other than fatigue for the surge nozzle. Corrosion degradation protection is provided by the combination of the austenitic stainless steel cladding of the surge nozzle inner radius and by the chemistry controls on the reactor coolant system. Strict chemistry standards are maintained to ensure a non-corrosive environment. Oxygen, chloride, fluoride and other contaminant concentrations are maintained below the thresholds known to be conducive to stress-corrosion cracking. Since the surge nozzle is cast, the typical failure mechanisms associated with weld material do not apply to this examination. Erosion and erosion/corrosion degradation is not credible at this location since the austenitic stainless steel cladding resists this mechanism. There is relatively low fluid velocity in the surge nozzle and reactor coolant chemistry minimizes the amount of particles in the fluid that could potentially cause erosion.

The licensee stated that, "fatigue degradation is a concern in the surge line nozzle area due to the potential thermal cycling caused by the insurge and outsurge of the reactor coolant flow. Since the surge nozzle is cast with the bottom head, there is no nozzle-to-vessel weld. The inner radius is believed to be less susceptible to fatigue problems than a nozzle to vessel weld. Initiation of fatigue cracking may have equal potential at the inner radius as compared to a nozzle-to-vessel weld, but the chances of having a pre-existing flaw are less likely in the inner radius casting than at a nozzle-to-vessel weld due to the manufacturing process. For this hypothetical flaw to be present, the licensee contended that it had to have been overlooked by the shop nondestructive examinations, which include surface, ultrasonic and radiographic examination. Inservice fatigue crack growth for such a flaw would be small since the pressurizer is hot during the insurges and outsurges resulting in relatively high fracture toughness of the material. A thermal sleeve installed in the surge nozzle provides a measure of protection from the affects of fluid temperature changes. Examinations are performed on the nozzle-to-safe end weld, which is within 18 inches of the inner radius."

The licensee has examined the nozzle-to-safe end weld satisfactorily without limitation. The nozzle-to-safe end weld is a stainless steel weld, not Alloy 82/182 material that is subject to primary water stress-corrosion cracking.

3.5 Licensee's Proposed Alternative

As an alternative to the volumetric (ultrasonic) examination as required by the applicable ASME Code, the licensee proposed to perform visual examination (VT-2) of the pressurizer surge nozzle area with the insulation installed in conjunction with the boric acid walk-down, performed every shutdown. Also, the nozzle area is included and documented in the Mode 3 walk-down of the reactor coolant system boundary, performed during each startup following refueling outages, as required by Table IWB-2500-1, Examination Category B-P, Item No. B15.10. Both of these activities are performed by qualified VT-2 examiners. These examinations are augmented by the leakage detection methods noted above. If the insulation is removed for maintenance or other purposes, the licensee intends to perform the ultrasonic examination of the inner radius section.

3.6 NRC Staff's Evaluation

The NRC staff evaluated the hardship in terms of radiological dose and examination issues. The NRC staff also evaluated the proposed visual examinations, and the potential causes of degradation and the structural integrity of the pressurizer surge line nozzle.

Hardship of ASME Code Compliance

In the relief request, the licensee discussed the dose estimates if the volumetric (ultrasonic) examinations were performed. The NRC staff asked the licensee to discuss how much dose was received during the heater sleeve repair in the eighth refueling outage (1R08) in 1991 and whether the current (2009) dose estimates agree with the dose received in outage 1R08. In the October 19, 2009, letter, the licensee responded that the actual dose received during the heater sleeve repair in 1991 is not known. However, the estimated dose for rework of two of the 78 heater cables was 1.2 rem. This was based on a dose rate of 600 millirem per hour (mR/hr) at that time, and the estimated time to disconnect (0.25 hours per cable) and reconnect (0.75 hours per cable) the two cables. The current (2009) dose rates are lower than the 1991 dose rates due to zinc addition, on-line chemistry, and chemical cleaning activities performed during refueling outages that have occurred since 1991.

The licensee stated further that the dose rates from the radiological survey taken during the seventeenth refueling outage (1R17) in 2006 are lower than the dose rates during the eighth refueling outage (1R08) in 1991 when the cable rework was performed. The current dose rates are considered to be stable at this time. The dose rates are based on the area where each worker will be positioned. The dose rate at the cables is 200 mR/hr; at the surge nozzle the rate is 100 mR/hr on contact and 60 mR/hr greater than 30 centimeters away. The electricians and insulators will be in direct contact with the cables and insulation above the cables. The boilermakers can stand further away while cleaning the area of interest. The examiners will be close to the nozzle area for the examination.

In the relief request, the licensee stated that the BVPS-2 pressurizer surge line was inspected ultrasonically during the last outage. The NRC staff asked the licensee to discuss how dose rates vary from BVPS-1 to BVPS-2 and what are the variances between the two units that allowed BVPS-2 to be examined, but not BVPS-1. In the October 19, 2009, letter, the licensee responded that a specific dose estimate was not developed for insulation removal and inspection of the BVPS-2 pressurizer surge line nozzle. However, the dose estimate for inspection of the BVPS-1 pressurizer surge line nozzle was substantial (17.7 rem), and the largest portion of the dose is attributed to disconnection and reconnection of heater cables (15.6 rem). The BVPS-1 and BVPS-2 pressurizer bottom heads have different insulation designs. The BVPS-2 insulation can be removed without disconnecting the heater cables because the insulation panels are split at the heater penetration center lines, allowing the panels to be lowered around the heater connections. This BVPS-2 design allows for the ultrasonic testing to be performed without hardship. The hardship of removing BVPS-1 insulation is discussed under the "Examination Issues" section below.

For the proposed visual examination, the licensee stated that the dose received during the VT-2 examination of the pressurizer surge line area is minimal (less than 1 millirem). Section XI of the ASME Code and site procedures allow a VT-2 to be performed without insulation removal by examining the horizontal surfaces of the insulation at the insulation joints and examining the surrounding area, including floor areas or equipment surfaces located underneath the pressurizer. Therefore, the time required to perform the VT-2 examination is a fraction of the time required for the ultrasonic examination, and the distance from the radiation source is much greater for the VT-2 examination compared to the ultrasonic (volumetric) examination.

It is evident that if a volumetric examination, such as ultrasonic testing (UT), were to be performed, the dose received by the personnel will be much higher than that of the visual examination.

Examination Issues

The licensee stated that the BVPS-1 surge nozzle inner radius section has not been examined. The BVPS-2 surge nozzle inner radius section and the surge nozzle-to-vessel weld were ultrasonically examined during the previous interval, and no recordable indications were found. Also, the BVPS-2 surge nozzle inner radius and the surge nozzle-to-vessel weld were ultrasonically examined during the first 10-year interval, and no recordable indications were found. The staff notes that even though the surge line nozzle at BVPS-2 has maintained its structural integrity without any flaws, this does not mean that BVPS-1 would maintain the same structural integrity. However, the inspection results for the surge line nozzle at BVPS-2 do provide a favorable reference for BVPS-1.

In the October 19, 2009, letter, the licensee discussed the difficulty of removing the insulation at the pressurizer bottom head. The BVPS-1 pressurizer bottom insulation was installed during plant construction prior to connecting the heater cables. Insulation panels span several heater pipes, and the holes provided in the insulation panels for the heater pipes do not allow the insulation panels to be lowered past the larger diameter of the ceramic terminal blocks (heater connections) at the ends of the heater pipes. Each rigid insulation panel spans several heater penetrations in all three circular rows of heaters, which prohibits panel removal past the heater connections.

In the relief request, the licensee stated that the adjacent nozzle-to-safe end weld, within 18 inches of the pressurizer surge line inner radius, had been examined. The staff asked the licensee to discuss whether there were any special considerations such as heater cable removal and insulation removal needed during the adjacent nozzle-to-safe end weld examination. In the October 19, 2009, letter, the licensee responded that the insulation covering the nozzle-to-safe end weld is separate from the pressurizer bottom head insulation. The surge nozzle-to-safe end weld is located at a sufficient distance from the pressurizer so that removal of insulation to access the weld does not require pressurizer heater cable disconnection or pressurizer bottom head insulation removal. Removal of the insulation covering the surge nozzle-to-safe end weld is not sufficient to allow inspection of the pressurizer surge line inner radius.

In the relief request, the licensee described the difficulty with installation of a remote camera to be used to constantly monitor the potential leakage at the surge line nozzle. In the October 19, 2009, letter, the licensee explained further that a flexible cable boroscope-type camera, containing a transmission and power cable for a portable light source was considered for this application. However, it would be difficult to lower the camera from the man-way (only access point), which is located off-center on the upper head. Because the man-way is not centered above the surge nozzle, it would be difficult to guide the camera through the approximate 2-foot diameter area at the center of the heaters. The camera would also need to be guided through two sets of baffle plates located inside the pressurizer that support the heaters and then through 3/8-inch diameter holes in the metal screen (retaining basket) located at the surge nozzle, which covers the area of interest. It is likely that the camera would have to be inserted through several different holes in the retaining basket to access the area of interest. To further complicate access, the holes in the screen are likely no longer 3/8 inch in diameter due to the forming process of the retaining basket, which makes the holes out of round. The surge nozzle contains a thermal sleeve that would obstruct access to the lower portion of the inner radius area. The thermal sleeve is located on the inside diameter of the nozzle and extends past the corner of the inner radius. The inner radius curves away from the thermal sleeve; however, the sleeve would significantly limit access to the lower portion of the radius.

The licensee had designed equipment to perform the required volumetric examination. In the October 19, 2009, letter, the licensee clarified that test UT transducer shoes designed for the specific geometry of the BVPS-1 surge nozzle were procured to facilitate volumetric examination of the nozzle. However, the dose associated with the examination preparations outweighs the benefits of performing the examination.

The NRC staff noted that the licensee requested relief from the requirements of Table IWB-2500-1, Examination Category B-D, Item No. B3.120. However, the applicable ASME Code, as stated by the licensee, ASME Code, Section XI, 2001 Edition, through 2003 Addenda, Table IWB-2500-1 does not contain this item number. In the October 19, 2009 letter, the licensee clarified that per 10 CFR 50.55a(b)(2)(xxi), Table IWB-2500-1 examination requirements, for licensees using the 1999 Addenda through the latest edition and addenda incorporated by reference in paragraph (b)(2) (FENOC uses the 2001 Edition through the

2003 Addenda for BVPS-1), the provisions of the 1998 Code Examination Category B-D, Item No. B3.120 must be applied. Therefore, Examination Category B-D and Item No. B3.120 were referenced in the relief request.

The NRC staff raised the concern regarding the impact of not performing the volumetric examination of the surge line nozzle on the leak-before-break analysis. In the October 19, 2009, letter, the licensee responded that the BVPS-1 leak-before-break analysis was performed with no credit or consideration of inservice inspection, and therefore, the inability to ultrasonically examine this location has no impact on the analysis.

The NRC staff finds that the licensee has addressed the difficulties with performing the required ultrasonic examination. The problem associated with the ultrasonic examination is that it is difficult to remove the bottom head insulation without incurring excessive radiological dose and the potential for damaging heater cable junctions.

Potential Causes of Degradation

The NRC staff notes that stress-corrosion cracking is caused by three parameters: corrosive environment, material susceptible to degradation, and unfavorable stresses. The surge piping experiences significant temperature differentials during operation because of functionality of the pressurizer. The temperature differential causes stresses on the pipe wall which over time may degrade the pipe. The staff asked the licensee to discuss whether any thermal transients that have occurred could have affected the structural integrity of the BVPS-1 pressurizer surge line nozzle. In the October 19, 2009, letter, the licensee responded that BVPS participated in the Westinghouse Owners Group efforts that led to the development of WCAP-14950, "Mitigation and Evaluation of Pressurizer Insurge/Outsurge Transients," dated February 1998. In addition, Westinghouse has performed site-specific analysis and described the analysis in WCAP-15351, "Evaluation of Pressurizer Transients Based on Plant Operations for Beaver Valley Unit 1," dated March 2000, for the pressurizer surge line nozzle and lower shell. That analysis incorporates a conservative representation of past transients, thermal stratification of the surge line during plant heat-up, shutdown and operation as well as a projection of past transients through the period of extended operation. WCAP-15351, Supplement 1, "Evaluation of Beaver Valley Unit 1 Pressurizer Insurge/Outsurge Transients with Revised Operating procedures," dated August 2005, led to modification of plant operating procedures to incorporate enhanced strategies for the mitigation of pressurizer insurges and outsurges.

The WCAP-15351, and WCAP-15351, Supplement 1 analyses, model the inner radius of the pressurizer surge line nozzle. The results of these analyses show that the inner radius of the nozzle is not the bounding location for either stress or cumulative usage factor. Therefore, the structural integrity of the pressurizer inner radius is not adversely affected by past thermal transients nor by postulated future thermal transients.

Another potential cause for pipe degradation is corrosion due to unfavorable water chemistry. The staff asked the licensee to discuss whether water chemistry transients that have occurred could have affected the pressurizer surge line nozzle. In the October 19, 2009, letter, the licensee responded that neither BVPS-1 nor BVPS-2 have experienced resin intrusions or chemical addition transients. Primary water chemistry is maintained to the Industry standard Electric Power Research Institute (EPRI) "PWR Primary Water Chemistry Guidelines." Therefore, there have been no water chemistry transients that affected the pressurizer surge line nozzle inner radius. The inner radius surface of the pressurizer and the pressurizer surge line are fabricated with austenitic stainless steel on all surfaces exposed to reactor coolant water (i.e., cladding). Stainless steel has been shown to be robust as a pressure boundary material and is resistant to chemical attack.

The NRC staff notes that austenitic stainless steel is resistant to chemical attack; however, stainless steel is not immune to stress-corrosion cracking. The NRC staff finds that the licensee has addressed the corrosion concern by following the EPRI guidance on PWR primary water chemistry and the thermal transient concern with stress analyses.

Structural Integrity of the Pressurizer Surge Nozzle

The licensee stated in the relief request that the inner radius of the pressurizer surge line is less susceptible to fatigue problems than a nozzle-to-vessel weld. In the October 19, 2009, letter, the licensee clarified that the statement of susceptibility to fatigue problems is in reference to the relative susceptibility of a pressurizer surge line nozzle that is constructed as a one-piece casting vs. a surge line nozzle that is constructed with a welded nozzle insert to the pressurizer bottom head shell. In a one-piece casting design, the surge line nozzle is integrally cast as part of the bottom head and there is no weld between the nozzle and the bottom head. The other design is to weld the surge nozzle to the bottom head shell.

If a nozzle is welded to a vessel shell such as the pressurizer, degradation may be initiated in the joint weld based on the industry's operating experience. For the one-piece casting design, the area of the inner radius does not have a weld susceptible to thermal fatigue from postulated reactor coolant insurge/outsurge events. The microstructure of the cast is unchanged in the area of the inner radius, thus eliminating the potential for fabrication defects that may be introduced during the welding process and microstructural differences that may be present at the casting to weld volume interface. These potential weld defects and microstructural discontinuities may provide stress risers for the initiation of a fatigue flaw. Therefore, the inner radius area of the nozzle in a single piece casting designed pressurizer bottom head would have a lower overall susceptibility to fatigue failure than one of welded design.

The staff finds in general, the one-piece casting nozzle would have less cracking problem than the nozzle that is welded to the vessel. However, the staff would not discredit the potential for cast components such as the surge line nozzle to contain fabrication defects during the manufacturing process. Also, there have been cases of cracking (fabrication and/or service induced) occurred in the stainless steel cladding that is welded to the vessel inside surface.

Proposed Alternative

The advantage of a volumetric examination is that it will detect flaws in the nozzle prior to the flaw becoming 100% through wall. The flaw may either be repaired or monitored closely. Visual examination of the subject nozzle that is covered with the insulation can only detect the flaw when the flaw has grown 100% through wall and the leak rate has become substantial that the leakage can be seen by the operators or detected by the leakage detection system.

The NRC staff raised a concern regarding how a visual examination can be identified and characterized an in-service flaw in the pressurizer surge line inner radius without a volumetric examination technique to ensure the integrity of the reactor coolant pressure boundary. The staff also questioned what measures the licensee would take to prevent a postulated in-service flaw from challenging the integrity of the reactor coolant pressure boundary. In the October 19, 2009, letter, the licensee responded that in the unlikely event that an in-service flaw should develop, it would be detected by the VT-2 examination. The licensee recognizes that inservice flaws cannot be identified or characterized on the inner radius without the volumetric examination.

The licensee stated further that the likelihood of a postulated fatigue flaw in a one-piece casting designed bottom head is considered small due to the lack of potential stress risers from fabrication induced defects and metallurgical discontinuities that may be present in the welded design. Also, plant procedures incorporate enhanced strategies for the mitigation of pressurizer insurges and outsurges, and primary water chemistry is maintained in accordance with EPRI guidelines. Therefore, the licensee contends that the proposed visual examinations and leakage monitoring program employed at BVPS-1 are appropriate to maintain a high confidence of continued safe operation.

The NRC staff finds that the licensee has presented a valid argument that the plant personnel would be subject to the unduly radiological dose should the volumetric (ultrasonic) examination be performed in accordance with the applicable ASME Code requirement.

The NRC staff finds that the surge line nozzle is not as susceptible to cracking because it is fabricated as part of the one-piece bottom head, although the one-piece construction does not preclude the potential for cracking. The licensee has followed the EPRI water chemistry guideline to minimize the potential for corrosion at the pressurizer surge line. Westinghouse analyses have analyzed thermal transients that would cause high stresses and showed that the surge line nozzle is not the bounding location for either stress or cumulative usage factor. The staff finds that the licensee has adequately addressed the concerns of susceptible material, corrosive environment, and unfavorable stresses.

The licensee has stated that it will perform the VT-2 examination with the insulation on the pipe during every shutdown. The licensee will also perform a walkdown to verify the absence of boric acid residue during every shutdown. The licensee will include the pressurizer surge line nozzle area in the Mode 3 walkdown of the reactor coolant system boundary during each startup. The NRC staff noted that the ASME Code required volumetric examination is performed once per 10-year ISI interval. The proposed visual examinations would be conducted about six times in the 10-year inspection interval, assuming a fuel cycle of 18 months.

On the basis of the NRC staff evaluation, the NRC staff finds that the proposed alternative provides a reasonable assurance that the structural integrity of the pressurizer surge line nozzle is maintained.

4.0 CONCLUSION

Based on the above evaluation, the NRC staff has concluded that compliance with the specified ASME Code would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety, and that the proposed visual examination, in combination with walkdowns during each refueling outage, provides reasonable assurance of the structural integrity of the pressurizer surge line nozzle. Therefore, pursuant to 10 CFR 50.55a(a)(3)(ii), the NRC staff authorizes the proposed alternative for BVPS-1 for the fourth 10-year ISI interval.

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

Principal Contributor: J. Tsao

Date: February 18, 2010

If you have any questions, please contact the Beaver Valley Project Manager, Nadiyah Morgan, at (301) 415-1016.

Sincerely,

Richard Guzman /RA/ for

Nancy L. Salgado, Chief
Plant Licensing Branch I-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-334

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NAME	JGall	NMorgan	SLittle	TChan	RGuzman for NSalgado
DATE	2/02/2010	2/02/2010	2/02/2010	12/18/09	2/18/2010

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