May 11, 2006

Mr. James A. Spina, Vice President
Calvert Cliffs Nuclear Power Plant, Inc.
Calvert Cliffs Nuclear Power Plant
1650 Calvert Cliffs Parkway
Lusby, MD 20657-4702

SUBJECT: CALVERT CLIFFS NUCLEAR POWER PLANT, UNIT NOS. 1 AND 2 - RELIEF REQUEST TO USE ALTERNATIVE TECHNIQUES FOR REPAIR OF WELDED NOZZLES (TAC NOS. MC9583 AND MC9584)

Dear Mr. Spina:

By letter dated December 21, 2005, as supplemented on February 23, 2006, Calvert Cliffs Nuclear Power Plant, Inc. (the licensee) submitted a relief request to use alternative repair and examination techniques for unacceptable indications in welded nozzles at Calvert Cliffs Nuclear Power Plant, Unit Nos. 1 and 2 (Calvert Cliffs Units 1 and 2). Specifically, the licensee requested relief from certain requirements of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code) concerning repair or replacement activities for pressure-retaining welds subject to Article IWA-4000 in Section XI. If leakage is identified during the pre-modification inspection of the Alloy 600 small-bore nozzles in the reactor coolant system (RCS) and a repair is required, the licensee proposed to repair the leaking nozzles using a half-nozzle repair technique based on Westinghouse Electric Company Report WCAP-15973-P, Revision 01, “Low-Alloy Steel Component Corrosion Analysis Supporting Small-Diameter Alloy 600/690 Nozzle Repair/Replacement Programs,” which was approved by the Nuclear Regulatory Commission (NRC) in a safety evaluation (SE) dated January 12, 2005. Therefore, the licensee has proposed, pursuant to Section 50.55a(a)(3)(i) of Part 50 of Title10 of the *Code of Federal Regulations* (10 CFR), the alternative repair and examination techniques.

The NRC staff has reviewed and evaluated the information regarding the relief request. The results are provided in the enclosed SE.

The NRC staff concludes that the proposed alternative for the repair and examination of the leaking RCS Alloy 600 nozzles using a half-nozzle design based on one of the techniques in Westinghouse Report WCAP-15973-P, Revision 01, provides an acceptable level of quality and safety. Therefore, the proposed alternative is authorized pursuant to 10 CFR 50.55a(a)(3)(i) for Calvert Cliffs Units 1 and 2 for the remainder of the third 10-year inservice inspection interval, which ends on June 30, 2009.

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

Sincerely,

/RA/

Richard J. Laufer, Chief
Plant Licensing Branch I-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket Nos. 50-317 and 50-318

Enclosure:
As stated

cc w/encl: See next page

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

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cc w/encl: See next page

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SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

REGARDING AN ALTERNATIVE FOR REPAIR AND EXAMINATION

OF REACTOR COOLANT SYSTEM HOT LEG INSTRUMENT NOZZLES

CALVERT CLIFFS NUCLEAR POWER PLANT, UNIT NOS. 1 AND 2

CALVERT CLIFFS NUCLEAR POWER PLANT, INC.

DOCKET NOS. 50-317 AND 50-318

1.0 INTRODUCTION

By letter dated December 21, 2005, as supplemented on February 23, 2006 (Agencywide Documents Access and Management System (ADAMS) Accession Nos. ML053620026 and ML060590049, respectively), Calvert Cliffs Nuclear Power Plant, Inc. (the licensee) submitted a relief request to use alternative repair and examination techniques for unacceptable indications in small-bore Alloy 600 welded nozzles at Calvert Cliffs Nuclear Power Plant, Unit Nos. 1 and 2 (Calvert Cliffs Unit Nos. 1 and 2). Specifically, the licensee requested relief from certain requirements of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code) concerning repair or replacement activities for pressure-retaining welds subject to Article IWA-4000 in Section XI. If leakage is identified during the pre-modification inspection of the small-bore nozzles welded to the reactor coolant system (RCS) hot legs and a repair is required, the licensee proposed to repair the leaking nozzles using a half-nozzle repair technique based on Westinghouse Electric Company Topical Report (TR) WCAP-15973-P, Revision 01, “Low-Alloy Steel Component Corrosion Analysis Supporting Small-Diameter Alloy 600/690 Nozzle Repair/Replacement Programs,” which was approved by the Nuclear Regulatory Commission (NRC) in a safety evaluation (SE) dated January 12, 2005. Therefore, the licensee has proposed, pursuant to Section 50.55a(a)(3)(i) to Part 50 of Title 10 of the *Code of Federal Regulations* (10 CFR), the alternative repair technique for the third 10-year inservice inspection (ISI) interval.

2.0 REGULATORY EVALUATION

The ISI of the ASME Code Class 1, 2, and 3 components are performed in accordance with Section XI of the ASME Code and the applicable addenda as required by 10 CFR 50.55a(g), except where relief has been granted by the Commission pursuant to 10 CFR 50.55a(g)(6)(i). 10 CFR 50.55a(a)(3) states that alternatives to the requirements of 50.55a(g) may be used, when authorized by the NRC, if (i) the proposed alternatives would provide an acceptable level of quality and safety or (ii) compliance with the specified requirements would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety.

Enclosure

Pursuant to 10 CFR 50.55a(g)(4), ASME Code Class 1, 2, and 3 components (including supports) shall 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 Inservice Inspection of Nuclear Power Plant Components," to the extent practical within the limitations of design, geometry, and materials of construction of the components. The regulations require that inservice examination of components and system pressure tests, including repair/replacement be conducted during the first 10-year interval and subsequent intervals and comply with the 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. The applicable code of ISI for Calvert Cliffs Unit Nos. 1 and 2 during the third 10-year ISI interval is the 1998 Edition (with no Addenda) of the ASME Code, Section XI.

3.0 TECHNICAL EVALUATION

3.1 System/Component for which Relief is Requested:

The proposed relief applies to the half-nozzle repair technique of the small-bore Alloy 600 nozzles welded to the RCS piping hot legs.

3.2 Code Requirements:

During the conduct of system pressure tests, any relevant condition identified during VT-2 visual examination as stated in subparagraph IWB-3522.1 of ASME Code, Section XI, shall require corrective action to meet the requirements of Paragraphs IWB-3142 and IWA-5250 prior to continued service. Paragraph IWA-5250 states that components requiring correction shall have repair/replacement activities performed in accordance with Article IWA-4000.

3.3 Proposed Alternative:

Any leaking nozzle will be modified by relocating the attachment weld from the interior surface of the pipe to the exterior surface of the pipe. The nozzle will be modified using the half-nozzle technique, where the outboard end of the Alloy 600 nozzle is removed by machining to approximately mid-wall of the hot leg piping. The outboard end of the nozzle is replaced with a short section (half-nozzle) of austenitic stainless steel attached with a partial penetration weld to the exterior surface of the pipe. The remainder of the Alloy 600 nozzle, including the original fabrication partial penetration weld, that contains the original flaw will remain in place.

3.4 Licensee's Basis for Relief

The NRC staff's SE dated January 12, 2005 on Westinghouse TR WCAP-15973-P, Revision 01, "Low-Alloy Steel Component Corrosion Analysis Supporting Small-Diameter Alloy 600/690 Nozzle Repair/Replacement programs," states that the report is acceptable for referencing in licensing applications in Combustion Engineering designed pressurized-water reactors to the extent specified and under the limitations delineated in the TR and in the SE. Sections 4.1, 4.2, and 4.3 of the SE present additional conditions to assess the applicability of the TR.

The TR evaluates the effect of component corrosion, resulting from primary coolant in the crevice region on component integrity, and the effects of propagation of flaws left in place by fatigue crack growth and stress-corrosion cracking mechanism. In the half-nozzle modification, small gaps of 1/8 inch or less remain between the remnants of the Alloy 600 nozzles and the new stainless steel nozzles. As a result, primary coolant (borated water) will fill the crevice between the nozzle and the wall of the pipe. Low alloy and carbon steels used for RCS components are clad with austenitic stainless steel to minimize corrosion resulting from exposure to primary coolant.

The licensee's plant-specific evaluation determined that the carbon and low alloy steel RCS components will not be unacceptably degraded by general corrosion as a result of the Alloy 600 half-nozzle replacement. Although some minor corrosion may occur in the crevice region of the replaced nozzles, the degradation will not proceed to the point where ASME Code requirements will be exceeded before the end of plant life including the period of extended operation. Postulated flaws were assessed for flaw growth and flaw stability in accordance with the ASME Code, Section XI, and the results demonstrated their compliance. Further, available laboratory data and field experience indicate that continued propagation of cracks into the carbon and low alloy steels by a stress-corrosion mechanism is unlikely.

3.4 Staff Evaluation

The licensee proposed to repair any leaking small-bore Alloy 600 nozzle in the RCS hot leg using one of the methods proposed in Westinghouse TR WCAP-15973-P, Revision 01, which the NRC staff approved in its SE dated January 12, 2005. The staff's review of the TR stated the following objectives with respect to implementing these repair/replacement methods:

1. Provides an acceptable method for calculating the overall general/crevice corrosion rate for the internal surfaces of the low-alloy or carbon steel materials that will now be exposed to the reactor coolant, and for calculating the amount of time the ferritic portions of the vessel or piping would be acceptable if corrosive wall thinning had occurred;

2. Provides an acceptable method of calculating the thermal-fatigue crack-growth life of existing flaws in the Alloy 600 nozzles and/or Alloy 82/182 weld material into the ferritic portion of the vessels or piping; and,

3. Provides acceptable bases and arguments for concluding that unacceptable growth of the existing flaw by stress corrosion into the vessels or piping is improbable.

Licensees seeking to use the repair/replacement methods of the TR are required to perform plant-specific evaluation of the effects of the above degradation modes in accordance with the methodology presented in the TR to confirm that the results of their evaluation are bounded under the TR and, therefore, will justify operation through the remaining licensed life of the facility. The results of the licensee's evaluations are as follows:

- The overall general corrosion rate for the Calvert Cliffs Unit Nos. 1 and 2 hot leg material based on estimated time spent at various modes of operation is 0.00074 inch per year.

- The maximum initial penetration bore diameter is 1.088 inches and the final penetration bore diameter after approximately 30 remaining years of operation is expected to be 1.132 inches.

- The maximum allowable penetration bore diameter due to general corrosion is 1.27 inches based on Reference 12 of WCAP-15973-P, Revision 01. The final calculated penetration bore diameter due to general corrosion is 1.132 inches. Based on the plant-specific corrosion rate of 0.00074 inch per year, it will take approximately 123 years of operation to reach the maximum allowable penetration bore diameter of 1.27 inches (limiting value). With a remaining operating life of 30 years, there is greater than a factor of 4 margin when actual plant operating time is compared against the time to reach the maximum allowable penetration bore diameter. Therefore, minor variances in start-up and shutdown conditions are not likely to change the margin that will result in exceeding the maximum allowable penetration bore diameter within the remainder of plant life.

- The geometry of the leaking penetration is bounded by that of the hot leg small-bore Alloy 600 J-groove weld geometry analyzed in the report from a final crack stability stand point.

- The plant-specific pressure-temperature profiles in the pressurizer water space for the limiting curves (cooldown curves) that were analyzed in the SE for the TR, bound the hot leg piping since the cooldown of the RCS, including the hot leg, is limited to 100 EF per hour by Technical Specifications. Further, the hot leg piping does not experience the same transients that the pressurizer experiences.

- For final crack stability evaluation of a postulated flaw associated with a nozzle in the hot leg piping, the linear elastic fracture mechanics (LEFM) analysis of RCS hot leg piping nozzles in Westinghouse Calculation Report CN-CI-02-71, Revision 1, was applicable to Calvert Cliffs Unit Nos. 1 and 2.

- An assessment of the effects of stress corrosion on the growth of existing flaws in the weld metal was based on reviews of RCS chemistry over the last three operating cycles. The application of hydrogen overpressure and control of impurities have successfully maintained the dissolved oxygen concentrations to less than 10 parts per billion (ppb) and the halide and sulfate ion concentrations to less than 150 ppb each.

Based on NRC staff's evaluation of WCAP-15973-P, Revision 01, the licensee has performed a general corrosion assessment, a thermal-fatigue crack growth assessment, and a stress-corrosion crack growth assessment of the proposed half-nozzle repair of the small-bore Alloy 600 nozzles welded to the RCS hot leg piping. In each of the above plant-specific assessments, the results are bounded by the assessment performed in the report. In addition, the staff determines that existing flaws in the weld metal are not likely to grow under stress corrosion. The NRC staff also finds LEFM analysis for the hot leg nozzle acceptable since it meets the ASME Code specified criteria with additional margins. Therefore, the NRC staff concludes that the results satisfy the conditions set forth in the SE dated January 12, 2005, in implementing the half-nozzle repair on hot leg small-bore nozzles.

The NRC staff also reviewed the welding procedure specification no. 55-WP1/8/F6AW1-008 that will be used to weld the replacement nozzles to the RCS hot leg pipe and found it to be acceptable because it meets the ASME Code. The licensee also has proposed to perform liquid penetrate tests at the half-thickness point of the weld and at the cover pass (full-thickness), supplemented by a VT-2 visual examination during the system leakage test of the replacement nozzles. The NRC staff, therefore, has determined that the proposed alternative repair would provide an acceptable level of quality and safety.

4.0 CONCLUSION

The NRC staff concludes that in implementing the half-nozzle repair technique as proposed by the licensee, it is acceptable to move the pressure boundary from inside surface to the outside surface of the pipe while leaving the material containing the original flaw inservice. The proposed repair is an alternative to the repair methodology of the applicable ASME Code, Section XI and is estimated to provide service through the remaining operating life of the facility. The proposed alternative repair would provide an acceptable level of quality and safety and is, therefore, authorized, pursuant to 10 CFR 50.55a(a)(3)(i), for the third 10-year ISI interval at Calvert Cliffs Unit Nos. 1 and 2. All other ASME Code, Section XI, requirements for which relief was not specifically requested and authorized herein by the NRC staff remain applicable, including a third party review by the Authorized Nuclear Inservice Inspector.

Principal Contributor: P. Patnaik

Date: May 11, 2006