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

December 5, 2008

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: REQUEST FOR ADDITIONAL INFORMATION RE: PROPOSED RELIEF REQUEST ISI-020 FOR REACTOR VESSEL WELD EXAMINATIONS - CALVERT CLIFFS NUCLEAR POWER PLANT, UNIT NO. 2 - (TAC NO. MD9773)

Dear Mr. Spina:

By letter dated October 1, 2008, Calvert Cliffs Nuclear Power Plant, Inc., the licensee for the Calvert Cliffs Nuclear Power Plant, Unit No. 2, submitted a proposed alternative from American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code), Section XI inservice inspection (ISI) requirements regarding examination of certain reactor pressure vessel welds. The licensee proposed an alternative ISI interval of 20 years in accordance with WCAP-16168-NP, Revision 2, "Risk-Informed Extension of the Reactor Vessel in-Service Inspection Interval."

The NRC staff has reviewed the information provided and has determined that additional information is needed to complete its review. Enclosed is the staff's request for additional information (RAI). As discussed with your staff, we understand that you intend to respond to this RAI by December 31, 2008.

Please contact me at 301-415-1364 if you have any questions.

Sincerely,

Douglas V. Pickett, Senior Project Manager
Plant Licensing Branch I-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-318

Enclosure:
As stated

cc w/encl: Distribution via Listserv

REQUEST FOR ADDITIONAL INFORMATION

CALVERT CLIFFS NUCLEAR POWER PLANT, UNIT NO. 2

PROPOSED RELIEF REQUEST ISI-020

REACTOR VESSEL WELD EXAMINATION EXTENSION

1. The Nuclear Regulatory Commission (NRC) staff seeks clarification of information regarding observed indications from recent ISI in Table 2 of Proposed Alternative ISI-020, Attachment (1). Provide a summary of each inspection report, including the initial inspection before being put into service, and a brief description of the procedure used at each inspection. Clearly state when the indications were found and if their discovery can be attributed to improved ultrasonic inspection procedures or if the indications were the result of aging mechanisms.

2. Identify source documents for determining the manganese content for the welds and plates in the reactor pressure vessel and describe how you averaged them when more than one data sources are available.

3. The application states that the applicable American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code) is the 1989 Edition, No Addenda. Considering the unique nature of the proposed alternative, which was intended to apply to the end of the current license, the NRC staff requests you revise the proposed alternative according to the following:

   - The edition and addenda of the ASME Code should be that which you would use for the fourth interval for other ASME Code applications.

   - The duration of the proposed alternative should end in 2036 when the current license expires; therefore, there will only be one additional scheduled ISI in 2019 covered by this request.

4. The staff notes the following:

   - The licensee's request, based on Topical Report (TR) WCAP-16168, "Risk-Informed Extension of the Reactor Vessel In-Service Inspection Interval," demonstrated that the $TWCF_{95\text{-}TOTAL}$ are orders of magnitude less than that for the bounding pilot plant vessel.

   - The calculation of $TWCF_{95\text{-}TOTAL}$ used the shift in the Charpy transition temperature produced by irradiation defined at the 30 ft-lb energy level, ΔT_{30}, calculated according to NUREG-1874, "Recommended Screening Limits for Pressurized Thermal Shock (PTS)."

   - Before the ΔT_{30} calculation in NUREG-1874 can be used for risk-informed decisions, the licensee should verify that the NUREG-1874 values of ΔT_{30} for the limiting baseline materials are comparable with the results from a plant-specific

or integrated surveillance program if the surveillance has been deemed consistent. The criteria set forth in proposed 10 CFR 50.61a, published on August 11, 2008 (73 FR 46557), paragraphs (f)(6)(i) through (f)(6)(iv) are to be implemented when three or more surveillance data points at different neutron fluences exist for the limiting beltline material.

Demonstrate, if applicable, using the criteria set forth in proposed 10 CFR 50.61a (f)(6)(i) through (f)(6)(iv) that the embrittlement model used in your application to calculate ΔT_{30} is applicable to the Calvert Cliffs Nuclear Power Plant, Unit No. 2.

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/RA/
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