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

August 6, 2010

Mr. George H. Gellrich, Vice President
Calvert Cliffs Nuclear Power Plant, LLC
Calvert Cliffs Nuclear Power Plant
1650 Calvert Cliffs Parkway
Lusby, MD 20657-4702

SUBJECT: REQUEST FOR ADDITIONAL INFORMATION RE: PRESSURIZER SAFETY VALVE TECHNICAL SPECIFICATION REVISION - CALVERT CLIFFS NUCLEAR POWER PLANT, UNIT NOS. 1 AND 2 - (TAC NOS. ME3348 AND ME3349)

Dear Mr. Gellrich:

By letter dated January 29, 2010 (Agencywide Document Access Management System (ADAMS) Accession No. ML100340586), and as supplemented on July 2, 2010 (ADAMS Accession No. ML101880024), Calvert Cliffs Nuclear Power Plant, LLC, the licensee, submitted a proposed change to Technical Specification (TS) 3.4.10, "Pressurizer Safety Valves," that would modify the existing Note within the TS. The Note allows the pressurizer safety valve lift settings to be outside the Limiting Condition for Operation limit as a result of temperature related lift setting drift, while the Unit is in applicable portions of Mode 3.

Based upon the Nuclear Regulatory Commission staff review, additional information will be necessary for the staff to complete its review. Enclosed is the staff's request for additional information (RAI). Based on discussions with your staff, we understand that you plan to respond to the enclosed RAI within 60 days of the date of this letter.

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

Sincerely,

Douglas V Pickett

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

Docket Nos. 50-317 and 50-318

Enclosure:
Request for Additional Information

cc w/encl: Distribution via Listserv

REQUEST FOR ADDITIONAL INFORMATION

PRESSURIZER SAFETY VALVE TECHNICAL SPECIFICATION REVISION

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

DOCKET NOS. 50-317 AND 50-318

By letter dated January 29, 2010, Calvert Cliffs Nuclear Power Plant, LLC, the licensee, submitted a proposed change to Technical Specification (TS) 3.4.10, "Pressurizer Safety Valves," that would modify the existing Note within the TS. By letter dated July 2, 2010, the licensee responded to a request for additional information (RAI) from the Nuclear Regulatory Commission staff. The following questions refer to the RAI responses.

A. RAI 1 Response

1. The response indicated that "evaluations" were made for the loss of normal feedwater (LNFW) and feedwater line break (FLB) events initiated from Mode 3 conditions.

   Discuss the methods for the "evaluations" and confirm that the methods used are NRC-approved methods or computer codes. In addition, please provide similar information for the control element assembly (CEA) ejection event, the CEA withdrawal event, and the excess load event.

2. The initial reactor coolant system (RCS) pressure of 700 psia was assumed for the evaluation of the LNFW and FLB events.

   For the initial RCS pressure of 700 psia assumed in the LNFW and FLB evaluation, please discuss compliance with Criterion 2 of 10 CFR 50.36(d)(2)(ii) that requires Technical Specification Limiting Conditions for Operation (LCOs) for "a process variable ...that is an initial condition of a design basis or transient analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier." It should be noted that the initial RCS pressure of 700 psia used in the evaluation is considered as a process variable referred in Criterion 2.

3. The RAI response stated that the LNFW and FLB transients were slow and that operators had sufficient time to implement emergency operating procedure (EOP) actions that would prevent the RCS pressure from reaching the pressurizer safety valve (PSV) opening setpoint.

   Discuss the results of the LNFW and FLB evaluation to show the transient times (i.e., mission times) at which the PSV would open for cases without operator actions, and discuss the EOP actions credited in the evaluation and demonstrate that the operators would reduce, in accordance with the EOP actions, the RCS pressure below the PSV setpoint within the calculated mission times. In addition, please provide similar information for the excess load event.

B. RAI 3 Response

1. The response indicated that the rate of change of power-high reactor protection system trip was used to terminate the CEA withdrawal event initiated from Mode 3, and that the event for Mode 3 was bounded by the event initiated from Mode 2, which relied on the variable high power trip for event termination.

   Discuss the allowable value for the rate of change of power-high trip and demonstrate that the trip on the signal from the rate of change of power-high trip for the Mode 3 event occurred sooner than that for the Mode 2 event, resulting in a lower RCS pressure during the Mode 3 event.

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Sincerely,
/RA/
Douglas V. Pickett, Senior Project Manager
Plant Licensing Branch I-1
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