George H. Gellrich
Vice President

Calvert Cliffs Nuclear Power Plant, LLC
1650 Calvert Cliffs Parkway
Lusby, Maryland 20657
410.495.5200
410.495.3500 Fax

CENG
a joint venture of
Constellation Energy EDF

CALVERT CLIFFS
NUCLEAR POWER PLANT

July 2, 2010

U. S. Nuclear Regulatory Commission
Washington, DC 20555

ATTENTION: Document Control Desk

SUBJECT: Calvert Cliffs Nuclear Power Plant
Unit Nos. 1 & 2; Docket Nos. 50-317 & 50-318
Response to Request for Additional Information – Pressurizer Safety Valve Technical Specification Revision

REFERENCES:

(a) Letter from Mr. G. H. Gellrich (CCNPP) to Document Control Desk (NRC) dated January 29, 2010, License Amendment Request: Pressurizer Safety Valve Technical Specification Revision

(b) Letter from Mr. D. V. Pickett (NRC) to Mr. G. H. Gellrich (CCNPP), dated May 7, 2010, 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)

In Reference (a), Calvert Cliffs Nuclear Power Plant, LLC. (Calvert Cliffs) submitted a license amendment request to the Nuclear Regulatory Commission (NRC) for a revision to Technical Specification 3.4.10, Pressurizer Safety Valves, for Calvert Cliffs Nuclear Power Plant, Units 1 and 2. In Reference (b), the NRC requested additional information be submitted to support their review of Reference (a). Attachment (1) provides the responses to the NRC's request for additional information contained in Reference (b).

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NRR

Should you have questions regarding this matter, please contact Mr. Douglas E. Lauver at (410) 495-5219.

Very truly yours,

STATE OF MARYLAND :
: **TO WIT:**
COUNTY OF CALVERT :

I, George H. Gellrich, being duly sworn, state that I am Vice President - Calvert Cliffs Nuclear Power Plant, LLC (CCNPP), and that I am duly authorized to execute and file this License Amendment Request on behalf of CCNPP. To the best of my knowledge and belief, the statements contained in this document are true and correct. To the extent that these statements are not based on my personal knowledge, they are based upon information provided by other CCNPP employees and/or consultants. Such information has been reviewed in accordance with company practice and I believe it to be reliable.

Subscribed and sworn before me, a Notary Public in and for the State of Maryland and County of St. Mary's, this 2ND day of July, 2010.

WITNESS my Hand and Notarial Seal:

Notary Public

My Commission Expires: March 1, 2011
Date

GHG/KLG/bjd

Attachment: (1) Response to Request for Additional Information – Pressurizer Safety Valve Technical Specification Revision

cc: D. V. Pickett, NRC
S. J. Collins, NRC
Resident Inspector, NRC
S. Gray, DNR

ATTACHMENT (1)

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION – PRESSURIZER SAFETY VALVE TECHNICAL SPECIFICATION REVISION

Calvert Cliffs Nuclear Power Plant, LLC
July 2, 2010

ATTACHMENT (1)

RESPONSES TO REQUEST FOR ADDITIONAL INFORMATION – PRESSURIZER SAFETY VALVE TECHNICAL SPECIFICATION REVISION

NRC RAI 1:

1. Updated Final Safety Analysis Report (UFSAR) Over-Pressure Transient Analysis

Chapter 14 of the Calvert Cliffs UFSAR (Revision 41) documents the results of the analyses of transients and accidents. The following three events initiating from hot-full power (HFP) relied on operation of the pressurizer safety valves (PSVs) to prevent the reactor coolant system (RCS) pressure from exceeding the reactor coolant pressure boundary (RCPB) limits:

1. *Loss of Load at HFP (UFSAR Figure 14.5-3)*
2. *Loss of Normal Feedwater (LNFW) at HFP (UFSAR Figure 14.6-3)*
3. *Feedwater Line Break (FLB) at HFP (UFSAR Figure 14.26-1)*

For operations other than MODE 1, Case 1, Loss of Load, listed above, is not a credible event because the turbine is off-line and the transient resulting from a turbine related fault cannot occur.

Discuss whether cases 2 and 3, LNFW and FLB, are credible events in MODE 3. If the events are credible, provide a qualitative and/or quantitative discussion to show that for Cases 2 and 3 above, an increase of the RCS pressure during transients will not reach the PSV set points specified in SR 3.4.10.1 for conditions within the applicable portions of MODE 3. If the PSV set points are exceeded, provide the results of analysis using Nuclear Regulatory Commission approved methods and PSV set points with maximum projected values of set point drift due to the temperature effects to show that the acceptable limits of the RCPB are met.

CCNPP RESPONSE TO RAI 1:

The PSVs lift setpoint is set at valve temperatures consistent with the temperatures the valve experiences at normal operating pressure and temperature. Due to the temperature related setpoint drift experienced by the PSV, the setpoint value will increase whenever the PSVs are subjected to lower ambient temperatures (as experienced during cooldowns from normal operating temperatures and during heat-ups from cold conditions) than those experienced at normal operating temperature. An evaluation was performed that used historical heat-up data to determine the expected pressurizer pressure that would exist when the lower valve temperature would cause its setpoint to exceed the Technical Specification limits. This value was approximately 600 psia. For the purposes of this evaluation a more conservative value (allows less margin to reach the PSV setpoint pressure) of 700 psia is used. It is also conservatively assumed that this pressure is reached five hours after shutdown.

To evaluate the LNFW, and FLB events in mode 3 the following assumptions were made;

1. The control rods are fully inserted and shutdown margin is being maintained using boron.
2. The primary system temperature is consistent with that which will produce a saturated pressure of greater than 700 psia in the pressurizer.
3. The secondary system is assumed to be providing sufficient steam and feed flow to maintain primary temperature and pressure.
4. No actuations, (such as atmospheric dump valve operation or pressurizer spray) that would mitigate these events were assumed to occur.
5. The main steam safety valves (MSSVs) are assumed operable per Technical Specifications 3.7.1 and a steam generator isolation signal per Technical Specification 3.3.4 is available.

6. Since the moderator temperature coefficient (MTC) can be either positive or negative during a fuel cycle depending on the time in a fuel cycle, the worst case condition of MTC for each event was selected.

Loss of Feedwater Flow

A Loss of Feedwater Flow event is a credible event during the applicable portions of Mode 3. It is defined as a reduction in feedwater flow to the steam generator (SG) without a corresponding reduction in steam flow from the SG. In Mode 3, the SGs may be fed using auxiliary feedwater with the main feedwater system shutdown. In this condition, a loss of auxiliary feedwater is assumed.

During this event, the pressure and temperature in the SG will begin to rise and the ability of the SG to remove heat from the RCS will lessen as inventory is boiled off. Reactor Coolant System temperature and pressure will begin to rise. In the presence of a positive MTC, positive reactivity will be inserted due to the RCS temperature and pressure increase. Secondary pressure will increase and a MSSV setpoint will be reached. Calvert Cliffs Technical Specification 3.7.1 requires that during Mode 3 while below 66% rated thermal power there will be at least 5 MSSVs operable per SG.

Lifting the MSSVs will deplete the SG inventory; however, due to maintaining subcritical conditions, a relatively long time is required to deplete the SGs. The transient is long enough to allow operators adequate time to implement emergency operating procedure actions that will mitigate the event and prevent reaching the PSV setpoint.

Feed Line Break

A FLB is a credible event in Mode 3. The RCS will respond in concert with the energy released from the secondary system. More energy released from the secondary system will cause the primary system temperature and pressure to decrease. Positive reactivity will be inserted if a large, negative MTC is assumed. If enough positive reactivity is inserted, core power will increase and temperature will rise.

The SG that is associated with the break in the feedwater line will dry out. The SGs will become isolated on steam generator isolation signal which will result in the pressure in the intact SG reaching the MSSV setpoint (at least five of which are required to be in service in Mode 3). The RCS will pressurize until the MSSVs lift, however the RCS pressure will remain below the PSV setpoint until after the intact SG dries out. Since a relatively long time is required to deplete the SG, sufficient time is available to allow operators to implement emergency operating procedure actions that will mitigate the event and prevent reaching the PSV setpoint.

NRC RAI 2:

2. Control Element Assembly (CEA) Ejection Analysis

As stated on page 3 of Attachment 1 to the January 29, 2010, license amendment application, the licensee stated there would be no challenge to PSV operation during a CEA ejection event while in the applicable portions of MODE 3. The licensee used the following statement in support of its position:

> *...the increase in RCS pressure during the event scenario is not addressed in any detail as the loss of RCS pressure barrier due to the CEA ejection prevents any significant RCS pressure buildup.*

Provide a discussion to show that the leakage in the reactor vessel head due to a CEA ejection is sufficiently large to remove the energy generated during a CEA ejection event such that the peak pressure does not reach the PSV set points specified in SR 3.4.10.1 for applicable portions of MODE 3.

CCNPP RESPONSE TO RAI 2:

A CEA Ejection event is discussed in Calvert Cliffs' UFSAR Section 14.13. The event is based upon the method described in Combustion Engineering topical report CENPD-190-A "C-E Method for Control Element Assembly Ejection Analysis," accepted per letter dated June 10, 1976, from Olan D. Parr (NRC) to A. E. Scherer (C-E). The methodology states that initiation of the event from zero power results in a larger change in the energy disposition to the reactor coolant than during full power initial conditions. The methodology assumes the CEA breaches the RCS by creating a 0.05 ft^2 break in the pressure boundary. This assumption is conservative relative to the CEAs' nozzle area of 0.09 ft^2.

Reactor coolant (RCS) pressure response during a CEA Ejection event was evaluated, where no credit is taken for any coolant flow through the break area, as well as taking no credit for either the pressurizer safety or relief valves' operation. Using these assumptions, RCS pressure increases to 2500 psia before RCS pressure starts to decrease. When the CEA Ejection event is analyzed with credit given for RCS leakage through the conservatively sized break area of 0.05 ft^2, RCS pressure initially increases but peaks out at 2450 psia. These bounding analyses demonstrate that in Mode 3 conditions the PSVs are not necessary to prevent exceeding RCS pressure safety limits (2750 psia) for a CEA Ejection event.

NRC RAI 3:

3. CEA Withdrawal Event and Excess Load Event

As stated on page 3 of Attachment 1 to the license amendment request dated January 29, 2010, the analyses presented in the UFSAR for the CEA withdrawal event and the excess load event assume hot zero power (Mode 2) conditions which are more bounding than Mode 3 conditions.

A CEA withdrawal event adds positive reactivity to the core, causing power to increase. As discussed in Section 14.2.2.1 of the UFSAR, the CEA withdrawal event initiating from Mode 2 conditions is terminated by a reactor trip from the variable high power trip (VHPT) signal. An excess load event decreases RCS temperature. In the presence of a negative moderator temperature coefficient, the decreased RCS temperature results in a positive reactivity addition, which in turn causes power to increase. As discussed in UFSAR Section 14.4.2.1, the excess load event initiating from Mode 2 conditions is also terminated by the VHPT signal.

The severity of the results of the analyses for these events in terms of maximum peak power and RCS pressure depends on the values assumed for the plant parameters such as: initial RCS temperature and pressure, reactivity insertion rate due to rod motion, excess steam release rate due to excess load, moderator and fuel temperature coefficients, pressure safety valves for overpressure protection, and available reactor trips for termination of the transient. It should be noted that the reactor trips for Mode 2 specified in the plant specific Technical Specifications may not be available for Mode 3.

Discuss the plant conditions and reactor protection system in Mode 3 to show that they are bounded by the Mode 2 conditions assumed in the UFSAR analyses for the CEA withdrawal event and the excess load event.

CCNPP RESPONSE TO RAI 3:

CEA Withdrawal

Technical Specification 3.3.2 states that four Rate of Change of Power-High Reactor Protective System bistable trip units, measurement channels, and automatic bypass removal features shall be operable in Modes 3, 4, and 5 when reactor trip circuit breakers are closed and any CEA is capable of being withdrawn. Thus if a CEA is capable of being withdrawn the automatic bypass removal feature is enabled leaving the Rate of Change of Power-High trip capable of terminating a power excursion.

Withdrawal of the CEA from subcritical conditions would cause a power rise that would be arrested by the Rate of Change of Power-High trip. The transient will be similar to, although less severe than, the sequence of events for the Mode 2 hot zero power event described in UFSAR Section 14.2.

Excess Load Event

Note – The assumptions listed in RAI#1 also apply to an Excess Load event.

For an Excess Load event, an increase in steam demand will cause SG pressures, RCS pressure, and RCS temperature to decrease. There will be a positive reactivity insertion if a large, negative MTC is assumed. If the reactivity insertion is sufficient to overcome the subcritical condition, then core power will increase.

For an Excess Load event, low SG pressure will trigger a main steam isolation signal that closes the main steam isolation valves. Technical Specification 3.7.2 requires that these valves be operable while the plant is in Mode 3 unless the valves are already closed. The closure of the main steam isolation valves blocks the egress of steam and causes an increase in SG pressure. The RCS temperature and pressure will rise and add negative reactivity due to the large, negative MTC. The primary and secondary system temperatures will rise until the MSSV setpoint is reached, at which time energy and inventory is released through the MSSVs. The transient is long enough to allow operators adequate time to implement emergency operating procedure actions that will mitigate the event and prevent reaching the PSV setpoint.