Calvert Cliffs Nuclear Power Plant

1650 Calvert Cliffs Parkway
Lusby, Maryland 20657

CENG
a joint venture of
Constellation Energy
eDF

CALVERT CLIFFS
NUCLEAR POWER PLANT

July 2, 2012

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
License Amendment Request: Revise Calculated Peak Containment Internal Pressure

In accordance with 10 CFR 50.90, Calvert Cliffs Nuclear Power Plant, LLC is submitting a request for an amendment to the Technical Specifications for Calvert Cliffs Nuclear Power Plant (CCNPP) Units 1 and 2. The proposed changes would revise the calculated peak containment internal pressure for the design basis loss-of-coolant accident (LOCA) described in Technical Specification 5.5.16 and the initial internal containment pressure limit in Technical Specification 3.6.4. The calculated peak containment internal pressure, P_a, is increased from 49.4 psig to 49.7 psig. This increase in P_a is due to an increase in the calculated mass and energy released into Containment during the blowdown phase of the design basis LOCA. In addition, a change to the containment pressure Technical Specification is needed to limit the upper bound initial pressure to less than or equal to 1.0 psig, from its current limit of less than or equal to 1.8 psig. This change is needed to support the calculation of peak containment internal pressure while maintaining the calculated peak containment internal pressure below the design limit of 50 psig.

Attachment (1) provides a description and assessment of the proposed changes. Attachment (2) provides the existing Technical Specification pages marked up to show the proposed changes. Calvert Cliffs Nuclear Power Plant requests approval of the proposed license amendment by June 30, 2013 with the amendment being implemented within 60 days.

In accordance with 10 CFR 50.91, a copy of this application, with attachments, is being provided to the designated Maryland Official.

There are no regulatory commitments contained in this letter.

A017
A002
NRR

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

Very truly yours,

Christopher R. Costanzo

Christopher R. Costanzo
Plant General Manager

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

I, Christopher R. Costanzo, being duly sworn, state that I am Plant General Manager - Calvert Cliffs Nuclear Power Plant, Inc. (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.

Christopher R Costanzo

Subscribed and sworn before me, a Notary Public in and for the State of Maryland and County of Calvert, this 2 day of July, 2012.

WITNESS my Hand and Notarial Seal:

Wendy L Hunter
Notary Public

Wendy L. Hunter
NOTARY PUBLIC
Calvert County, Maryland
My Commission Expires 1/9/2014

My Commission Expires: 1/9/14

Date

GHG/PSF/bjd

Attachments: (1) Description and Assessment of Proposed Changes
(2) Proposed Technical Specification Changes

cc: N. S. Morgan, NRC
W. M. Dean, NRC
Resident Inspector, NRC
S. Gray, DNR

ATTACHMENT (1)

DESCRIPTION AND ASSESSMENT OF PROPOSED CHANGES

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1.0 SUMMARY DESCRIPTION

This letter is a request for an amendment to Renewed Operating Licenses DPR-53 and DPR-69 for Calvert Cliffs Nuclear Power Plant (Calvert Cliffs), Units 1 and 2, to change the calculated peak containment internal pressure for the design basis loss-of-coolant accident (LOCA) described in Technical Specification Section 5.5.16, "Containment Leakage Rate Testing Program." The calculated peak containment internal pressure for this event, P_a, would be increased from 49.4 psig to 49.7 psig. This increase in P_a is due to an increase in the calculated mass and energy released into containment during the blowdown phase in the LOCA containment response analysis. Technical Specification 3.6.4, "Containment Pressure," requires a conforming change to the upper limit allowed for the containment pressure during plant operations to match the initial conditions assumed in the accident analyses which determined the proposed P_a.

2.0 DETAILED DESCRIPTION

Peak containment internal pressure during a LOCA is calculated to determine, in part, whether the design pressure limit for the containment building would be exceeded during a design basis accident. The peak containment pressure for a LOCA is presented in Technical Specification 5.5.16, "Containment Leakage Rate Testing Program." A LOCA is initiated by the rupture of the primary coolant system piping. The primary coolant flashes to steam and escapes through the pipe break. As the steam is released to Containment, containment atmosphere pressure and temperature quickly increase. The structures in Containment will absorb energy and condense steam, counteracting the initial pressure and temperature increase. The containment air coolers and containment spray system, which are activated by the increase in containment pressure, act to reduce containment pressure and temperature by removing energy from the containment atmosphere as the event progresses. During a LOCA event, the initial blowdown of the primary coolant system adds mass and energy to the containment atmosphere.

Mass and energy release data was provided by Westinghouse as an input to the LOCA containment response analysis. These mass and energy release values are used to calculate P_a using the GOTHIC computer code.

While performing an Extended Power Uprate study for another customer, Westinghouse identified a non-conservative LOCA mass and energy release input for the containment response analysis. Westinghouse determined that the mass and energy generated by the thermal hydraulic response computer code (CEFLASH-4A) during the blowdown phase of the event was not adequately detailed, with respect to time step data, during the early stages of the transient for use in downstream containment response calculations. The error occurred due to an incorrect setting in the time step resolution of the CEFLASH-4A computer code for the blowdown phase of the LOCA analysis. This time step value is established by the code user and could impact calculations at other plants if these calculations were performed with similar time step settings. In fact, Reference 1 documents a similar Technical Specification change approved for the Palisades Nuclear Power Plant due to this error in time step resolution.

This error resulted in an under-prediction of the mass and energy released to the Containment during the blowdown phase of the event. The inclusion of the additional mass and energy release led to an increase in the calculated peak containment internal pressure. The calculated peak containment pressure remains below the containment design pressure of 50 psig.

The under-prediction of mass and energy released into Containment occurred only for the LOCA event. The mass and energy release assumed in the main steam line break analysis was not affected.

Technical Specification Section 5.5.16, "Containment Leakage Rate Testing Program", currently states:

> "The peak calculated containment internal pressure for the design basis loss-of-coolant accident, P_a, is 49.4 psig. The containment design pressure is 50 psig."

The proposed change would revise Technical Specification 5.5.16 by replacing the P_a value of 49.4 psig with a value of 49.7 psig. The revised Technical Specification Section 5.5.16 would read as follows:

> "The peak calculated containment internal pressure for the design basis loss of coolant accident, P_a, is 49.7 psig. The containment design pressure is 50 psig."

An input into the calculation of P_a is the initial containment pressure at the time the LOCA occurs. The P_a calculation assumes that the containment is at the peak pressure allowed by Technical Specification 3.6.4, "Containment Pressure." Since inclusion of the additional mass and energy release led to an increase in the calculated peak containment internal pressure, a reduction in the initial containment pressure assumed was necessary to prevent exceeding the containment design pressure. Therefore, Technical Specification 3.6.4, "Containment Pressure," requires a conforming change to the upper limit allowed for the containment pressure during plant operations to match the initial conditions assumed in the accident analyses which determined the proposed P_a.

In addition, Technical Specification 3.6.4, Containment Pressure, currently states:

> "Containment pressure shall be $\geq$ -1.0 psig and $\leq$ 1.8 psig."

The proposed change would revise the upper bound of containment pressure to 1.0 psig. The revised Technical Specification 3.6.4 would state:

> "Containment pressure shall be $\geq$ -1.0 psig and $\leq$ 1.0 psig."

3.0 TECHNICAL EVALUATION

Technical Specification Section 5.5.16, "Containment Leakage Rate Testing Program," describes the calculated peak containment internal pressure during the design basis LOCA, P_a. The containment leak rate testing program uses P_a when leak testing Containment, containment isolation valves, and containment penetrations, including the containment airlock, in accordance with 10 CFR Part 50 Appendix J.

Technical Specification 5.5.16 specifies that the containment leakage rate testing program shall be in accordance with the guidelines of Regulatory Guide 1.163, "Performance-Based Containment Leak-Test Program," dated September 1995. This Regulatory Guide references Nuclear Energy Institute 94-01, Revision 0, dated July 26, 1995, "Industry Guideline for Implementing Performance Based Option of 10 CFR 50 Appendix J." This document provides methods acceptable to the Nuclear Regulatory Commission (NRC) Staff for complying with the provisions of Option B in Appendix J to 10 CFR Part 50. Nuclear Energy Institute 94-01 references American National Standards Institute (ANSI) 56.8-1994, "Containment System Leakage Testing Requirements," for detailed descriptions of the technical methods and techniques for performing Type A, B, and C tests.

Upon NRC approval, this increase in P_a would be reflected in the 10 CFR Part 50, Appendix J containment leak rate testing procedures. The procedures that govern 10 CFR Part 50, Appendix J, Type A testing will be changed to reflect the revised calculation peak containment internal pressure in Technical Specification 5.5.16. Title 10 CFR Part 50, Appendix J, Type A testing is required by ANSI 56.8-1994 to be performed at a pressure not less than 0.96 times the calculated peak containment pressure (P_a) and not more than the containment design pressure. These values will be updated in the procedure prior to the next performance of the Type A test for either Unit.

Title 10 CFR Part 50, Appendix J, Type B and C test procedures do not require revision upon approval of the proposed license amendment. The applicable ANSI Standard (56.8-1994) requires that Type B and C testing be performed at a pressure not less than P_a (except for airlock door seals, which may have a lower pressure specified) and not more than 1.1 times P_a when a higher differential pressure results in increased sealing. Site procedures for Type B and C testing require that testing be performed within a range of pressures that, with the revised P_a, will continue to be within the range of pressures required by ANSI 56.8-1994. Therefore, the Type B and C test procedures will not require revision to remain valid.

This change in P_a does not affect the offsite radiological consequences of a LOCA as previously analyzed in the Updated Final Safety Analysis Report. The LOCA offsite radiological dose consequence analysis is based on the maximum allowable containment leakage rate of 0.16% of containment air weight per day. The analysis assumes that containment leakage during the first 24 hours of the event is 0.16% of containment atmosphere by weight and 0.08% of containment atmosphere by weight afterward. It also assumes that the release of radionuclides to Containment is instantaneously mixed with containment air within the containment free air volume. Since the maximum allowable containment leakage rate is not being revised, containment leakage assumed in the LOCA analysis is not impacted. Therefore, the increase in the calculated peak containment internal pressure does not impact the offsite radiological consequences of the LOCA accident analysis.

The change in P_a does not affect the analysis of radiological consequences of a LOCA with respect to radiological dose to the Control Room operators. Calculated Control Room operator dose during a LOCA is dependent on the maximum allowable containment atmosphere leakage rate and is unaffected by calculated peak containment internal pressure, as discussed above. Since the maximum allowable containment leakage rate is not being revised, dose to the Control Room operators is not affected by a change in peak containment pressure.

The change in P_a does not adversely affect environmentally qualified equipment within containment. A review of the effects of this change in mass and energy release on the environmental qualification of equipment in containment determined that the equipment remained qualified for service in the revised pressure and temperature environment.

As noted earlier, another input into the calculation of P_a is the initial containment pressure at the time the LOCA occurs. The calculation assumes that the containment is at the peak pressure allowed by Technical Specification 3.6.4, "Containment Pressure." Since inclusion of the additional mass and energy release led to an increase in the calculated peak containment internal pressure, a reduction in the initial containment pressure assumed was necessary to prevent exceeding the containment design pressure. Therefore, Technical Specification 3.6.4, "Containment Pressure," requires a conforming change to the upper limit allowed for the containment pressure during plant operations to match the initial conditions assumed in the accident analyses which determined the proposed P_a.

4.0 REGULATORY SAFETY ANALYSIS

4.1 Applicable Regulatory Requirements/Criteria

No regulation specifically addresses the determination of the mass and energy release into the containment following a postulated design basis accident. The General Design Criteria 16 and 50 address the capability of the containment to withstand the containment pressure resulting from a postulated design basis LOCA.

General Design Criterion 16, "Containment Design," states that reactor containment and associated systems shall be provided to establish an essentially leak-tight barrier against the uncontrolled release of radioactivity to the environment and to assure that the containment design conditions important to safety are not exceeded for as long as postulated accident conditions require.

General Design Criteria 50, "Containment Design Basis," states that the reactor containment structure, including access openings, penetrations, and the containment heat removal system shall be designed so that the containment structure and its internal compartments can accommodate, without exceeding the design leakage rate and with sufficient margin, the calculated pressure and temperature conditions resulting from any loss-of-coolant accident. This margin shall reflect consideration of (1) the effects of potential energy sources which have not been included in the determination of the peak conditions, such as energy in steam generators and, as required by § 50.44, energy from metal-water and other chemical reactions that may result from degradation but not total failure of emergency core cooling functioning, (2) the limited experience and experimental data available for defining accident phenomena and containment responses, and (3) the conservatism of the calculational model and input parameters.

Additionally, the regulation (10 CFR Part 50, Appendix J, Option B) defines P_a as the calculated peak containment internal pressure related to the design basis LOCA as specified in the Technical Specification and specifies the requirements for containment leakage rate testing. The requirements of Technical Specification 5.5.16, "Containment Leakage Rate Testing Program," provide more detailed leakage rate testing requirements.

Although Calvert Cliffs licensing basis is the draft General Design Criteria, these General Design Criteria continue to be met following the change in calculated peak containment pressure. The environmental qualification of equipment within containment is not affected by the change in peak calculated containment pressure following a LOCA. The change in calculated peak containment pressure will be reflected in future 10 CFR Part 50 Appendix J, Type A containment integrated leakrate testing, so containment integrity is not impacted by the change. The change in calculated peak containment pressure does not impact the maximum allowable containment leakage rate and therefore does not impact control room operator dose. The calculated peak containment pressure remains below containment design pressure.

Although Calvert Cliffs licensing basis is the draft General Design Criteria, these General Design Criteria also continue to be met for the conforming change necessary in the containment initial pressure limit. The environmental qualification of equipment within containment is not affected by the change in the initial pressure limit. The change in the initial containment pressure limit helps to preserve the maximum allowable containment leakage rate (by preserving the calculated peak pressure) and therefore does not impact Control Room operator dose. The containment initial pressure limit change is necessary to ensure that the calculated peak containment pressure remains below containment design pressure.

4.2 Precedent

This proposed license amendment request is similar to one approved for Palisades on January 19, 2012 (Reference 1). The Palisades license amendment did not include a conforming change to the initial containment pressure, however, the change to their P_a Technical Specification is essentially the same as the change to P_a requested here.

4.3 No Significant Hazards Consideration

Calvert Cliffs Nuclear Power Plant is proposing a license amendment to Technical Specification Section 5.5.16, "Containment Leakage Rate Testing Program." The proposed amendment would increase the calculated peak containment internal pressure for the design basis loss of coolant accident, P_a, from 49.4 psig to 49.7 psig. In addition, a change to Technical Specification 3.6.4, "Containment Pressure," is needed to limit the upper bound initial containment pressure to less than or equal to 1.0 psig, from its current limit of less than or equal to 1.8 psig. This change is needed to support the calculation of peak containment internal pressure while maintaining the calculated peak containment internal pressure below the design limit of 50 psig.

Calvert Cliffs has evaluated whether or not a significant hazards consideration is involved with the proposed amendment by focusing on the three standards set forth in 10 CFR 50.92, as discussed below:

1. *Does the proposed change involve a significant increase in the probability or consequences of an accident previously evaluated?*

   Response: No.

   The proposed change to P_a and the initial containment pressure limit does not alter the assumed initiators to any analyzed event. The probability of an accident previously evaluated will not be increased by this proposed change. The change in P_a and the initial containment pressure limit will not affect radiological dose consequence analyses. The radiological dose consequence analyses assume a certain containment atmosphere leak rate based on the maximum allowable containment leakage rate, which is not affected by the change in calculated peak containment internal pressure. The 10 CFR Part 50, Appendix J containment leak rate testing program will continue to ensure that containment leakage remains within the leakage assumed in the offsite dose consequence analyses. The consequences of an accident previously evaluated will not be increased by this proposed change.

   Therefore, operation of the facility in accordance with the proposed change to Technical Specification Sections 3.6.4 and 5.5.16 will not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. *Does the proposed change create the possibility of a new or different kind of accident from any accident previously evaluated?*

   Response: No.

   The proposed change provides a higher P_a than currently described in the Technical Specification. This change is a result of an increase in the mass and energy release input for the loss of coolant accident containment response analysis. The calculated peak containment pressure remains below the containment design pressure of 50 psig because of the change in the initial containment pressure limit, which is an initial condition of the peak pressure calculation. This change does not involve any alteration in the plant configuration (no new or different type of equipment will be installed) or make changes in the methods governing normal plant operation. The change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

   Therefore, operation of the facility in accordance with the proposed change to Technical Specification Sections 3.6.4 and 5.5.16 would not create the possibility of a new or different kind of accident from any previously evaluated.

3. *Does the proposed change involve a significant reduction in a margin of safety?*

   Response: No.

   The calculated peak containment pressure remains below the containment design pressure of 50 psig. Since the radiological consequence analyses are based on the maximum allowable containment leakage rate, which is not being revised, the change in the calculated peak containment pressure does not represent a significant change in the margin of safety.

   Therefore, operation of the facility in accordance with the proposed change to Technical Specification Sections 3.6.4 and 5.5.16 does not involve a significant reduction in the margin of safety.

4.4 Conclusions

Calvert Cliffs has determined that the proposed change does not require any exemptions or relief from regulatory requirements, other than the Technical Specifications, and does not affect conformance with any regulatory requirements or criteria.

Based on the considerations above, (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will continue to be conducted in accordance with the site licensing basis, and (3) the approval of the proposed change will not be inimical to the common defense and security or to the health and safety of the public.

5.0 ENVIRONMENTAL CONSIDERATION

The proposed amendment would change a requirement with respect to installed facility components located within the restricted area of the plant as defined in 10 CFR Part 20. However, the proposed amendment does not involve (i) a significant hazards consideration, (ii) a significant change in the types or significant increase in the amounts of any effluent that may be released offsite, or (iii) a significant increase in individual or cumulative occupational radiation exposure. Accordingly, the proposed amendment meets the eligibility criterion for categorical exclusion set forth in 10 CFR 51.22(c)(9). Therefore, pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the proposed amendment.

6.0 REFERENCE

1. Letter from M. L. Chawla (NRC) to Vice President, Operations (Palisades), dated January 19, 2012, Issuance of Amendment re: Revise Calculated Peak Containment Internal Pressure

ATTACHMENT (2)

PROPOSED TECHNICAL SPECIFICATION CHANGES

3.6 CONTAINMENT SYSTEMS

3.6.4 Containment Pressure

LCO 3.6.4 Containment pressure shall be $\geq$ -1.0 psig and $\leq$ 1.~~8~~ 0 psig.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Containment pressure not within limits.	A.1 Restore containment pressure to within limits.	1 hour
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3.	6 hours
	<u>AND</u>	
	B.2 Be in MODE 5.	36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.4.1	Verify containment pressure is within limits.	12 hours

5.5 Programs and Manuals

5.5.16 Containment Leakage Rate Testing Program

A program shall be established to implement the leakage testing of the containment as required by 10 CFR 50.54(o) and 10 CFR Part 50, Appendix J, Option B. This program shall be in accordance with the guidelines contained in Regulatory Guide 1.163, "Performance-Based Containment Leak-Test Program," dated September 1995, including errata, as modified by the following exceptions:

a. Nuclear Energy Institute (NEI) 94-01 – 1995, Section 9.2.3: The first Unit 1 Type A test performed after the June 15, 1992 Type A test shall be performed no later than June 14, 2007. The first Unit 2 Type A test performed after the May 2, 2001 Type A test shall be performed no later than May 1, 2016.

b. Unit 1 is excepted from post-modification integrated leakage rate testing requirements associated with steam generator replacement.

c. Unit 2 is excepted from post-modification integrated leakage rate testing requirements associated with steam generator replacement.

The peak calculated containment internal pressure for the design basis loss-of-coolant accident, P_a, is 49.~~4~~7 psig. The containment design pressure is 50 psig.

The maximum allowable containment leakage rate, L_a, shall be 0.16 percent of containment air weight per day at P_a.

Leakage rate acceptance criteria are:

a. Containment leakage rate acceptance criterion is $\leq 1.0\ L_a$. During the first unit startup following testing, in accordance with this program, the leakage rate acceptance criterion are $\leq 0.60\ L_a$ for Types B and C tests and $\leq 0.75\ L_a$ for Type A tests.