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

December 30, 2015

Mr. Bryan C. Hanson
President and Chief Nuclear Officer
Exelon Nuclear
4300 Winfield Road
Warrenville, IL 60555

SUBJECT: CALVERT CLIFFS NUCLEAR POWER PLANT, UNIT NOS. 1 AND 2 - ISSUANCE OF AMENDMENT RE: REVISION TO PRESSURIZER SAFETY VALVE TECHNICAL SPECIFICATIONS (CAC NOS. MF3541 AND MF3542)

Dear Mr. Hanson:

The U.S. Nuclear Regulatory Commission (NRC) has issued the enclosed Amendment No. 315 to Renewed Facility Operating License No. DPR-53, and Amendment No. 293 to Renewed Facility Operating License No. DPR-69 for the Calvert Cliffs Nuclear Power Plant, Unit Nos. 1 and 2, respectively. These amendments consist of changes to the technical specifications (TSs) in response to your application dated February 13, 2014 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML14050A374), as supplemented by letter dated June 22, 2015 (ADAMS Accession No. ML15177A115).

These amendments revise TS 3.4.10, "Pressurizer Safety Valves," to modify as-found lift tolerances in the Surveillance Requirement (SR). The changes to the SR reduce the lift setpoint for valve RC-201, and increase the allowable as-found setpoint tolerance on valves RC-200 and RC-201.

A copy of the related Safety Evaluation is enclosed. A Notice of Issuance will be included in the Commission's next regular biweekly *Federal Register* notice.

Sincerely,

Alexander N. Chereskin, Project Manager
Plant Licensing Branch I-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket Nos. 50-317 and 50-318

Enclosures:
1. Amendment No. 315 to DPR-53
2. Amendment No. 293 to DPR-69
3. Safety Evaluation

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

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

CALVERT CLIFFS NUCLEAR POWER PLANT, UNIT 1

CALVERT CLIFFS NUCLEAR POWER PLANT, LLC

EXELON GENERATION COMPANY, LLC

DOCKET NO. 50-317

Amendment No. 315
Renewed License No. DPR-53

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by Exelon Generation Company, LLC (Exelon, the licensee), dated February 13, 2014, as supplemented by letter dated June 22, 2015, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C.(2) of Renewed Facility Operating License No. DPR-53 is hereby amended to read as follows:

(2) Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 315, are hereby incorporated into this license. Exelon Generation shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of the date of its issuance and shall be implemented at or before the end of second refueling outage following approval of this amendment.

FOR THE NUCLEAR REGULATORY COMMISSION

Travis L. Tate

Travis Tate, Chief
Plant Licensing Branch I-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment:
Changes to the License and Technical Specifications

Date of Issuance: December 30, 2015

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

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

CALVERT CLIFFS NUCLEAR POWER PLANT, UNIT 2

CALVERT CLIFFS NUCLEAR POWER PLANT, LLC

EXELON GENERATION COMPANY, LLC

DOCKET NO. 50-318

Amendment No. 293
Renewed License No. DPR-69

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by Exelon Generation Company, LLC (Exelon, the licensee), dated February 13, 2014, as supplemented by letter dated June 22, 2015, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C.(2) of Renewed Facility Operating License No. DPR-69 is hereby amended to read as follows:

(2) Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 293, are hereby incorporated into this license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of the date of its issuance and shall be implemented at or before the end of second refueling outage following approval of this amendment.

FOR THE NUCLEAR REGULATORY COMMISSION

Travis L. Tate

Travis Tate, Chief
Plant Licensing Branch I-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment:
Changes to the License and
Technical Specifications

Date of Issuance: December 30, 2015

ATTACHMENT TO LICENSE AMENDMENTS

AMENDMENT NO. 315 TO RENEWED FACILITY OPERATING LICENSE NO. DPR-53

AMENDMENT NO. 293 TO RENEWED FACILITY OPERATING LICENSE NO. DPR-69

DOCKET NOS. 50-317 AND 50-318

Replace the following pages of the Renewed Facility Operating Licenses with the attached revised pages. The revised pages are identified by amendment number and contain marginal lines indicating the areas of change.

Remove Pages	Insert Pages
3	3

Replace the following pages of the Appendix A Technical Specifications with the attached revised pages. The revised pages are identified by amendment number and contain marginal lines indicating the areas of change.

Remove Pages	Insert Pages
3.4.10-2	3.4.10-2

(4) Exelon Generation pursuant to the Act and 10 CFR Parts 30, 40, and 70, to receive, possess, and use, in amounts as required, any byproduct, source, and special nuclear material without restriction to chemical or physical form, for sample analysis or instrument calibration or associated with radioactive apparatus or components; and

(5) Exelon Generation pursuant to the Act and 10 CFR Parts 30 and 70 to possess, but not separate, such byproduct and special nuclear materials as may be produced by the operation of the facility.

C. This license is deemed to contain and is subject to the conditions set forth in 10 CFR Chapter I and is subject to all applicable provisions of the Act, and the rules, regulations, and orders of the Commission, now or hereafter applicable; and is subject to the additional conditions specified and incorporated below:

(1) Maximum Power Level

Exelon Generation is authorized to operate the facility at steady-state reactor core power levels not in excess of 2737 megawatts-thermal in accordance with the conditions specified herein.

(2) Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 315, are hereby incorporated into this license. Exelon Generation shall operate the facility in accordance with the Technical Specifications.

(a) For Surveillance Requirements (SRs) that are new, in Amendment 227 to Facility Operating License No. DPR-53, the first performance is due at the end of the first surveillance interval that begins at implementation of Amendment 227. For SRs that existed prior to Amendment 227, including SRs with modified acceptance criteria and SRs whose frequency of performance is being extended, the first performance is due at the end of the first surveillance interval that begins on the date the Surveillance was last performed prior to implementation of Amendment 227.

(3) Additional Conditions

The Additional Conditions contained in Appendix C as revised through Amendment No. 305 are hereby incorporated into this license. Exelon Generation shall operate the facility in accordance with the Additional Conditions.

(4) Secondary Water Chemistry Monitoring Program

Exelon Generation shall implement a secondary water chemistry monitoring program to inhibit steam generator tube degradation. This program shall include:

(4) Exelon Generation pursuant to the Act and 10 CFR Parts 30, 40, and 70, to receive, possess, and use, in amounts as required, any byproduct, source, and special nuclear material without restriction to chemical or physical form, for sample analysis or instrument calibration or associated with radioactive apparatus or components; and

(5) Exelon Generation pursuant to the Act and 10 CFR Parts 30 and 70 to possess, but not separate, such byproduct and special nuclear materials as may be produced by the operation of the facility.

C. This license is deemed to contain and is subject to the conditions set forth in 10 CFR Chapter I and is subject to all applicable provisions of the Act, and the rules, regulations, and orders of the Commission, now and hereafter applicable; and is subject to the additional conditions specified and incorporated below:

(1) Maximum Power Level

Exelon Generation is authorized to operate the facility at reactor steady-state core power levels not in excess of 2737 megawatts-thermal in accordance with the conditions specified herein.

(2) Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 293 are hereby incorporated into this license. The licensee shall operate the facility in accordance with the Technical Specifications.

(a) For Surveillance Requirements (SRs) that are new, in Amendment 201 to Facility Operating License No. DPR-69, the first performance is due at the end of the first surveillance interval that begins at implementation of Amendment 201. For SRs that existed prior to Amendment 201, including SRs with modified acceptance criteria and SRs whose frequency of performance is being extended, the first performance is due at the end of the first surveillance interval that begins on the date the Surveillance was last performed prior to implementation of Amendment 201.

(3) Less Than Four Pump Operation

The licensee shall not operate the reactor at power levels in excess of five (5) percent of rated thermal power with less than four (4) reactor coolant pumps in operation. This condition shall remain in effect until the licensee has submitted safety analyses for less than four pump operation, and approval for such operation has been granted by the Commission by amendment of this license.

(4) Environmental Monitoring Program

If harmful effects or evidence of irreversible damage are detected by the biological monitoring program, hydrological monitoring program, and the

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
B. Required Action and associated Completion Time not met. OR Two pressurizer safety valves inoperable.	B.1	Be in MODE 3.	6 hours
	AND		
	B.2	Reduce all RCS cold leg temperatures to ≤ 365°F (Unit 1), ≤ 301°F (Unit 2).	12 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.10.1	Verify each pressurizer safety valve is OPERABLE in accordance with the Inservice Testing Program. The lift settings shall be within limits as specified below:	In accordance with the Inservice Testing Program

Valve	As Found Lift Setting (psia)	As Left Lift Setting (psia)
RC-200	≥ 2475 and ≤ 2575	≥ 2475 and ≤ 2525
RC-201	≥ 2475 and ≤ 2600	≥ 2500 and ≤ 2550

UNITED STATES
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SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO REVISION OF PRESSURIZER SAFETY VALVE

TECHNICAL SPECIFICATIONS

AMENDMENT NO. 315 TO RENEWED FACILITY OPERATING LICENSE NO. DPR-53

AMENDMENT NO. 293 TO RENEWED FACILITY OPERATING LICENSE NO. DPR-69

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

EXELON GENERATION COMPANY, LLC

DOCKET NOS. 50-317 AND 50-318

1.0 INTRODUCTION

By application dated February 13, 2014 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML14050A374), as supplemented by letter dated June 22, 2015 (ADAMS Accession No. ML15177A115), Exelon Generation Company, LLC (the licensee) requested an amendment to the licenses for Calvert Cliffs Nuclear Power Plant, Units 1 and 2 (CCNPP), DPR-53 and DPR-69, to revise the technical specifications (TSs) for the pressurizer safety valve setpoints. The supplemental letter dated June 22, 2015, provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the U.S. Nuclear Regulatory Commission (NRC) staff's original proposed no significant hazards consideration determination as published in the *Federal Register* (FR) on July 22, 2014 (79 FR 42549).

The license amendment request (LAR) proposed changes to TS 3.4.10, "Pressurizer Safety Valves." The proposed changes would modify the as-found lift tolerances in the surveillance requirement (SR) for the pressurizer safety valves (PSVs) and reduce the lift setpoint for valve RC-201 from 17,686 kilopascal (kPa) (2565 pounds per square inch absolute (psia)) to 17,409 kPa (2525 psia). The nominal setpoint for valve RC-200 would remain at the design pressure 17,237 kPa (2500 psia). The proposed change increases the allowable PSV setpoint tolerance on valve RC-200 from -1%/+2% to -1%/+3% and on RC-201 from ±2% to -2%/+3%. The as-left tolerances would remain at ±1%.

The licensee requested these changes to reduce an unnecessarily restrictive SR. In support of these changes, the licensee submitted an evaluation of the effect of the changes on the plant design basis and the results of re-analyses of four design-basis transients that are impacted by the proposed changes.

Enclosure 3

2.0 REGULATORY EVALUATION

The construction permits for CCNPP were issued by the Atomic Energy Commission (AEC) on July 7, 1969, and the operating licenses were issued on July 31, 1974, for Unit No.1 and August 13, 1976, for Unit No.2. The AEC published the final rule that added 10 CFR Part 50, Appendix A, "General Design Criteria [GDC] for Nuclear Power Plants," in the *Federal Register* (36 FR 3255) on February 20, 1971, with the rule becoming effective on May 21, 1971. As stated in SECY-92-223, dated September 18, 1992, the Commission decided not to apply the Appendix A GDC to plants with construction permits issued prior to May 21, 1971. The CCNPP Updated Final Safety Analysis report (UFSAR), Revision 47, dated August 27, 2014, states that the plant was designed and constructed to meet the intent of the GDC published in July 1967. The plant's GDC are discussed in the UFSAR, Appendix 1C, "AEC Proposed General Design Criteria for Nuclear Power Plants."

2.1 Component Description

The function of the pressurizer is to maintain the reactor pressure within limits appropriate for safe operation, including assuring that the maximum pressure does not exceed safety limits. CCNPP includes two PSVs for each unit, whose main purpose is to provide Reactor Coolant System (RCS) overpressure protection. The PSVs are set to open at approximately 2500 psia, and at approximately 2565 psia, to ensure that the RCS pressure safety limit of 2750 psia is not exceeded during design basis accidents (DBAs). The licensee submitted the LAR to demonstrate that the proposed modifications to the TSs are such that the applicable regulatory requirements listed below are satisfied following implementation of the TS changes. The LAR includes transient analysis that demonstrates the capability of the CCNPP reactor protective features to terminate limiting events and mitigate their consequences without exceeding the reactor coolant pressure boundary (RCPB) safety limit.

2.2 Regulatory Requirements

The regulatory requirements and guidance documents which the NRC staff used in the review of the application are listed below:

- Section 50.36 of 10 CFR, "Technical specifications," provides the regulatory requirements for the content required in the TSs. As stated in 10 CFR 50.36(c)(1)(i)(A), safety limits for nuclear reactors are limits upon important process variables that are found to be necessary to reasonably protect the integrity of certain of the physical barriers that guard against the uncontrolled release of radioactivity.

- Section 50.36(c)(2)(ii) states that a TS limiting condition for operation of a nuclear reactor must be established for each item meeting one or more of the listed criteria. Pressurizer safety valves satisfy 10 CFR 50.36(c)(2)(ii), Criterion 3, which states "A structure, system, or component that is part of the primary success path and which functions or actuates to mitigate a design basis accident or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier."

- NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition," (SRP) Section 5.2.2, Revision 3, "Overpressure Protection," provides guidance to determine whether the systems that provide overpressure protection to the RCPB satisfies the requirements of GDCs 15 "Reactor Coolant System Design" and 31 "Fracture Prevention of Reactor Coolant Pressure Boundary," and will perform its intended functions during all plant operating and accident conditions. Even though NUREG-0800 was used in the NRC staff evaluation, the licensee is not held to GDC 15 and GDC 31 as Calvert Cliffs is licensed to the draft GDC, as noted above in Section 2.0.

- NUREG-1432, "Standard Technical Specifications, Combustion Engineering Plants," Revision 4.0, April 2012.

- Draft GDC 9, "Reactor Coolant Pressure Boundary (Category A)," requires that the RCPB has an "...exceedingly low probability of gross rupture or significant leakage throughout its design lifetime."

- Draft GDC 33, "Reactor Coolant Pressure Boundary Capability (Category A)," requires that the RCPB will not rupture under static and dynamic loads placed on any RCPB component due to any sudden and inadvertent release of energy to the coolant.

- American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code), Section III, Article NB-7000, "Overpressure Protection." Article NB-7311, Relieving Capacity of Pressure Relief Devices, specifies that the overpressure protection system provide sufficient relief capacity to prevent a pressure increase greater than 10% above the RCPB design pressure, accounting for loses through piping and other components. This requirement is also stated in NUREG-0800.

3.0 TECHNICAL EVALUATION

3.1 Background

Over the last several years, the licensee has submitted several licensee event reports (LERs) related to PSV setpoints at CCNPP. In LER 317/2014-003, the licensee described how both PSVs in Unit No. 1 were found to have setpoints below the lower tolerance, as a result of faulty lift-spring assemblies. LER 317/2014-003 described previous events at both units, dating back to 2008, which involved setpoint testing abnormalities (low and high) due to normal or excessive drift. Based on this experience, the licensee submitted the subject LAR seeking to change the tolerance band for the PSVs. In LER 317/2014-003, the licensee also stated that they were awaiting approval of a LAR to change the setpoint tolerance.

The PSVs at CCNPP are manufactured by Dresser Consolidated. There have been a variety of events associated with safety valve setpoints in nuclear power plants and other industries. For those events that did not involve obvious design or manufacturing flaws, the corrective actions usually involved procedural changes to better account for the nature of the setpoint drift and environmental conditions. The licensee consulted with the manufacturer during their evaluation of the setpoint drift to determine the best corrective action.

Both units at CCNPP are pressurized water reactors designed by Combustion Engineering. The primary coolant system is protected from overpressure events by two spring-loaded, back-pressure compensated, totally enclosed safety valves on the pressurizer. The self-actuated PSVs discharge steam from the pressurizer to a quench tank inside the containment building. The PSVs, in conjunction with the steam generator relief valves in the secondary system, and the reactor protection and safeguards systems will protect the primary system against overpressure in the event of a complete loss of heat sink.

As per the CCNPP TSs, both PSVs are required to be operable during plant operation in Modes 1 and 2 and in Mode 3 when temperatures are greater than 185°C (365°F) for Unit 1, and greater than 149°C (301°F) for Unit 2. Below those temperatures in Mode 3 and while in Modes 4, 5, and 6, the two PSVs are not required to be operable as overpressure protection is maintained through the low temperature over pressure protection covered by TS 3.4.12.

The CCNPP design basis includes (1) the maximum transient pressure allowable in the RCS pressure vessel in the ASME Code, Section III, Article NB-7000, is 110% of the design pressure; and (2) the maximum transient pressure allowable in the reactor coolant system (RCS) piping, valves, and fittings under the ASME 1967 Standard Code for Piping, B31.7, is 110% of design pressure. The as-left pressure limits are based on the ±1% tolerance requirement in Article NB-7000 for lifting pressures above 6,895 kPa (1000 psig). PSV operability ensures that the RCPB and RCS piping, valves and fittings will not exceed 110% of design pressure. The design pressure is 17,237 kPa (2500 psia) and the limit of 18,961 kPa (2750 psia) is contained in TS 2.1.2.

3.2 Staff Evaluation

3.2.1 Structural Integrity of Piping and Pipe Supports

For PSV RC-200 the licensee proposed a one percent increase in the as-found high setpoint pressure from 2,550 psia to 2,575 psia. The NRC staff evaluated this change and found that the one percent increase in the as-found tolerance high setpoint for PSV RC-200 is not considered significant enough to affect the structural integrity of piping and pipe supports. After reviewing the licensee's evaluation, the staff has concluded that the design basis load stresses will remain less than the current design basis allowable stresses. In addition, the licensee reviewed the associated pipe support restraint loads and deflections due to the increase in PSV RC-200 setpoint tolerance against the current restraint design. The licensee determined that the restraints will accommodate the effects of the PSV RC-200 setpoint tolerance increase. Therefore, the staff concludes that there is reasonable assurance that this change will not adversely affect the structural integrity of the PSV discharge piping and associated pipe supports.

3.2.2 Accident Analysis and Methodology

The licensee proposed changes to the PSV setpoints and surveillance tolerances are as follows:

Valve	As-Found Lift Setpoint (kPa/psia)	As-Left Lift Setpoint (kPa/psia)
	Current Setpoints	
RC-200	≥ 17,065 and ≤ 17,582 ≥ 2475 and ≤ 2550	≥ 17,065 and ≤ 17,410 ≥ 2475 and ≤ 2525
RC-201	17,333 ≥ and ≤ 18,036 ≥ 2514 and ≤ 2616	≥ 17,513 and ≤ 17,858 ≥ 2540 and ≤ 2590
	Proposed Setpoints	
RC-200	≥ 17,065 and ≤ 17,754 ≥ 2475 and ≤ 2575	≥ 17,065 and ≤ 17,410 ≥ 2475 and ≤ 2525
RC-201	≥ 17,065 and ≤ 17,927 ≥ 2475 and ≤ 2600	≥ 17,237 and ≤ 17,582 ≥ 2500 and ≤ 2550

The proposed changes to the CCNPP TSs only involve the setpoint values. The form and style of TS 3.4.10 would continue to conform to the model specification in NUREG-1432 for Combustion Engineering plants.

To demonstrate RCPB integrity, the licensee used the S-RELAP5 methodology found in the document listed in TS 5.6.5.b.8, AREVA document EMF-2310(P)(A), Revision 1, "SRP Chapter 15 Non-LOCA [Loss of Coolant Accident] Methodology for Pressurized Water Reactors". This document describes the S-RELAP5 code, its uses, and any limitations and conditions on the use of the S-RELAP5 code. The S-RELAP5 code is a computer code that is used for light water reactor transient analysis. In the NRC staff's safety evaluation (SE) for EMF-2310(P)(A), Table 1 lists the transients that the S-RELAP5 code may be used to analyze. These transients include the four events evaluated by the licensee in this LAR.

The licensee requested to adopt the use of EMF-2310(P)(A) in its LAR dated November 23, 2009. In its SE dated February 18, 2011, the NRC staff approved this document for use at CCNPP and added a license condition to the Appendix C of the CCNPP Facility Operating License (FOL) to restrict the use of the S-RELAP5 code. This license condition includes a restriction to the use of this methodology that requires prior transient-specific NRC approval to analyze performance relative to RCPB pressure integrity until NRC approval is obtained for a generic or CCNPP specific basis. The license condition was added in order to capture the overpressure aspects of limiting pressure transients that were not analyzed at the time of the amendment that adopted EMF-2310(P)(A) into the TSs.

Specifically, the S-RELAP5 code is only approved for this one transient-specific application of the methodology to CCNPP as described in this SE, and is not a generic approval of the methodology. The confirmatory calculations described below provide the basis for the NRC approval of this specific application of the S-RELAP5 code.

The licensee evaluated all of the DBAs and transients, as described in Chapter 14 of the UFSAR, against the PSV setpoint changes. The changes in PSV setpoint tolerance will only affect analyses which have a primary pressure excursion large enough that the PSVs open. The licensee determined that there are three events addressed in the SRP and UFSAR required to be analyzed for primary and secondary overpressure:

- Loss of External Electrical Load (SRP 15.2.1/UFSAR 14.5)
- Loss of Normal Feedwater Flow (SRP 15.2.7/UFSAR 14.6)
- Feedwater System Pipe Break (SRP 15.2.8/UFSAR 14.26)

In addition, the Control Element Assembly (CEA) Ejection (SRP 15.4.8/UFSAR 14.13) is analyzed only for peak primary pressure, since the event does not challenge the secondary system pressure limit.

For all of the other DBAs and transients addressed in the SRP and UFSAR, the licensee determined that the proposed changes are bounded by the existing design basis analysis (analysis of record) or bounded by the re-analysis of the four listed events, as is described in detail in the LAR.

For each transient event analysis, the nodalization, chosen parameters, conservative input, and sensitivity studies were reviewed for applicability to the PSV setpoint tolerance change in compliance with EMF-2310(P)(A). The nodalization used for the calculations supporting the PSV setpoint tolerance change is specific to CCNPP, in accordance with EMF-2310(P)(A), with renodalization between the pressurizer and the PSVs. In addition, there is a significantly large length of PSV inlet piping at CCNPP, which was specifically modeled.

The process variables used by AREVA in event analyses were biased to assure conservative results. AREVA biased the process variables consistent with the SRP guidelines and EMF-2130(P)(A). When a TS limit was a parameter to be biased, the allowed operating range and uncertainty for the power level being considered was conservatively bounded. Other process variables that do not have a TS limit, but may significantly affect the results of the transient calculations, were also biased to bound allowed operating ranges and uncertainties. The input conditions for the re-analyses are generally consistent with the existing analysis of record, except where operating experience has provided a basis for a more accurate parameter value. In addition, the licensee clarified that the range over which the parameter was varied included the uncertainty and control deadband, as applicable, and the limiting cases represent the process parameter biasing that most adversely affects the over-pressure results.

The S-RELAP5 non-LOCA model relies on user-specified input for fuel thermal properties for the core heat structures. The core heat structures are used to determine fuel temperature for Doppler feedback and average core fuel surface heat flux. For the CEA Ejection event, the fuel thermal conductivity input for both the heat structures conservatively accounted for degradation with exposure. Thermal conductivity degradation is an important parameter for the CEA Ejection event, since it is partially mitigated by Doppler feedback, whereas the other overpressure events do not have a significant increase in fission power and are therefore less affected.

All of the design-basis transients that require PSV actuation assume operation of both valves to limit RCS pressure, opening at the high range of the as-found setpoint. Single failure of a safety valve is not assumed in the design-basis events, nor required by the ASME Code.

The peak primary system pressure following the most severe anticipated transient, the feedwater line break, is 18,823 kPa (2730 psia) which is less than the 18,961 kPa (2750 psia) ASME Code allowable (110 percent of the design pressure) with no credit taken for nonsafety-grade relief systems. The NRC staff reviewed the AREVA user specified inputs to the S-RELAP5 code and determined that they are conservative for the purpose of calculating peak RCS pressure.

An on-site audit was conducted on March 25, 2015, as described in the audit plan sent to the licensee by letter dated May 26, 2015 (ML15083A018). During the audit, the licensee responded to questions regarding the LAR and allowed the NRC staff and contractors to review related records.

A request for additional information (RAI) was sent to the licensee by letter dated May 6, 2015 (ML15112A374) to clarify certain details of the amendment request and information discussed during the audit. The licensee responded to the RAI in a letter dated June 22, 2015 (ML15177A115). In the response, the licensee explained how the revised PSV as-found setpoints were selected (1) to be consistent with general industry practice for comparability, (2) to ensure that the peak system pressure for the limiting event would not exceed the ASME limit of 110% design pressure, and (3) ensure sufficient operating margin to avoid inadvertent PSV actuation.

Confirmatory analyses were performed by the NRC staff using the TRACE code to validate the licensee's analysis results. The NRC's TRACE code is an advanced computational thermal-hydraulic code that is able to analyze LOCAs and other system transients in both pressurized- and boiling-water reactors. The confirmatory analyses accounted for clarifying details provided in the licensee's RAI responses, particularly for modelling and sensitivity studies. The confirmatory analyses demonstrate that the licensee's AREVA analyses are conservative. Attachment 3 to the LAR provided a summary of the supporting AREVA analyses. The AREVA analyses determined a main feedwater line break is the limiting pressurization event and is described in the AREVA main feedwater line break calculation.

AREVA made modifications to their base CCNPP S-RELAP5 model in order to obtain a bounding calculation for the main feedwater line break pressurization analysis. The acceptance criteria to be demonstrated is that the peak primary system pressure remains below 110% of the design pressure, i.e. remain below 2750 psia. The location of the peak pressure is the bottom of the reactor pressure vessel (RPV). The AREVA analysis for the main feedwater line break determined that the peak pressure of 2730.7 psia occurred for a break size of 0.02 square feet (sq. ft.). Parameter biasing in the AREVA analysis was mainly responsible for determining the limiting break size that would produce the maximum peak pressure.

The NRC staff modified the CCNPP TRACE model following the input specifications in the TRACE V5.0 User's Manual, Volume 1: Input Specification. These changes were made to simulate the feedwater line break transient based on changes in the AREVA S-RELAP5 model that are described in the AREVA feedwater line break calculation.

3.3 Technical Evaluation Conclusions

A plot of peak pressure at the bottom of the RPV and pressure at the safety valve are shown for the limiting case in Figure 1. There are two local peaks in the pressure for each case, one for small breaks and one for larger breaks. Based on the results of the confirmatory calculations the NRC staff made the following observations:

1. The peak pressures for the confirmatory analysis are significantly lower than the peak pressures reported by AREVA. This may be due to conservatisms that were not readily apparent from the AREVA documentation, and were therefore not included in the TRACE model. Both the TRACE code and the base CCNPP input model are best estimate.

2. The TRACE calculations are in agreement with the S-RELAP5 calculations in terms of the break sizes that yield local peak pressures, namely 0.02 sq. ft. and 0.34 sq. ft.

3. The double peak nature of the pressure response and the small difference between the peak pressures explains how a significant difference in limiting break size can occur.

4. The TRACE code analyses performed by the staff confirm that the peak pressurization results presented in the AREVA feedwater line break calculation are conservative and provide adequate support for the subject LAR.

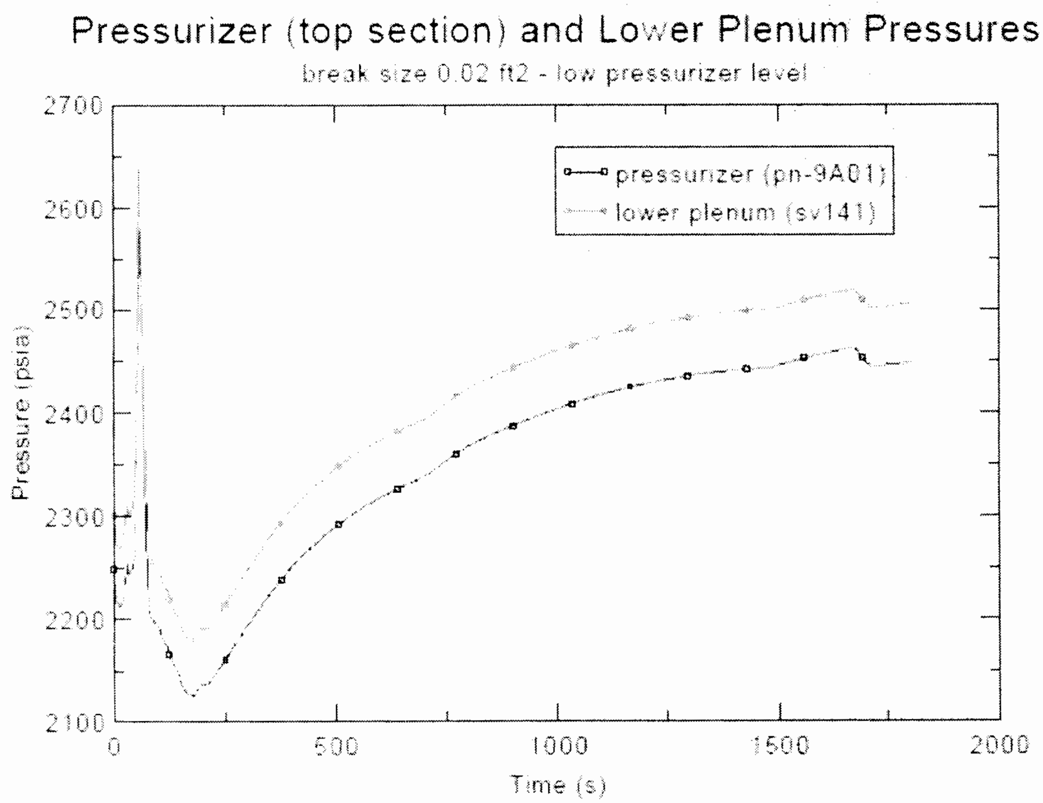

Figure 1 TRACE Pressure Response for 0.02 sq. ft. FWL Break - level biased low

As explained above, the TRACE confirmatory analyses demonstrate that the licensee's analyses to support the proposed changes in the as-found PSV lift setpoints are conservative. Moreover, calculations with the TRACE code confirmed that the peak system pressure for the limiting feedwater line break event will not exceed 110% of the design pressure, in accordance with the ASME Code, Section III, Article NB-7000.

Based on the NRC staff's evaluation of the LAR, in conjunction with findings from the confirmatory analyses it performed, and the licensee's choice and conservative biasing of the input parameters to the S-RELAP5 code, the NRC staff concludes that the licensee has properly implemented the conditions in EMF-2310(P)(A), and has correctly identified and reanalyzed the affected design-basis transients.

In addition, based on the NRC staff's confirmatory analyses, the NRC staff concludes that the licensee is allowed to use the transient-specific application of the S-RELAP5 code for this specific application only.

The overpressure protection system will continue to provide sufficient relief capacity with the proposed tolerance bands and prevent a pressure increase greater than 10% above the RCPB design pressure, accounting for loses through piping and other components. The proposed changes, therefore, satisfy the requirements of draft GDC 9 and draft GDC 33 that the overpressure protection system maintain RCS pressure within acceptable design limits and with sufficient margins during normal operation and anticipated operational occurrences. Accordingly, the NRC staff concludes that since the licensee has met the applicable regulatory requirements discussed above, the transient-specific changes to the S-RELAP5 analyses and the changes to TS 3.4.10 are acceptable.

4.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Maryland State official was notified of the proposed issuance of the amendments. The State official had no comments.

5.0 ENVIRONMENTAL CONSIDERATION

The amendments change a requirement with respect to the installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. The NRC staff has determined that the amendments involve no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendments involve no significant hazards consideration, and there has been no public comment on such finding published in the FR on July 22, 2014 (79 FR 42549). Accordingly, the amendments meet the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendments.

6.0 CONCLUSION

The Commission has concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) there is reasonable assurance that such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendments will not be inimical to the common defense and security or to the health and safety of the public.

7.0 REFERENCES

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2. Letter from George Gellrich to U.S. Nuclear Regulatory Commission, "Request for Additional Information Regarding the As-found Lift Tolerances for the Pressurizer Safety Valves License Amendment Request," June 22, 2015 (ADAMS Accession No. ML15177A115).

3. U.S. Nuclear Regulatory Commission, "Biweekly Notice; Applications and Amendments to Facility Operating Licenses and Combined Licenses Involving No Significant Hazards Considerations," *Federal Register*, Vol. 79, No. 140, July 22, 2014, pp. 42549.

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13. American Society of Mechanical Engineers, Boiler and Pressure Vessel Code, Section III, Article NB-7000, "Overpressure Protection," July 1, 2006.

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15. EMF-2310(P)(A), Revision 1, "SRP Chapter 15 Non-LOCA Methodology for Pressurized Water Reactors." (ADAMS Accession No. ML041810031)

16. Stuart A. Richards, USNRC, Letter to Mallay, James F., Framatome ANP, Richland, Inc., Acceptance for Referencing of Licensing Topical Report EMF-2310(P), Revision 0, "SRP Chapter 15 Non-LOCA Methodology for Pressurized Water Reactors (TAC No. MA7192)," dated May 11, 2011, ADAMS Accession No. ML011310533.

17. Letter from Thomas E. Trepanier (CCNPP, LLC) to USNRC Document Control Desk, "License Amendment Request – Transition from Westinghouse Nuclear Fuel to AREVA Nuclear Fuel," dated November 23, 2009, ADAMS Accession No. ML093350099.

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22. Chereskin, Alexander N., U.S. Nuclear Regulatory Commission, Letter to Gellrich, George H., Exelon Generation Company, LLC, "Calvert Cliffs Nuclear Power Plant Unit Nos. 1 and 2 – Request for Additional Information Regarding the As-Found Lift Tolerances for the Pressurizer Safety Valves License Amendment Request (TAC Nos. MF3541 and MF3542)," dated May 6, 2015 (ADAMS Accession No. ML15112A374).

23. TRACE Calculation Notebook, Dan Prelewicz and Shawn Marshall, December 2010 (proprietary).

24. CCNPP PSV Setpoint Tolerance – Feedwater Line Break (FLB) Primary System Overpressure, AREVA Calculation 32-9187689-001, Revision 0, Revision 001, March 4, 2013 (Proprietary).

Principal Contributors: Fred Forsaty

Date: December 30, 2015

A copy of the related Safety Evaluation is enclosed. A Notice of Issuance will be included in the Commission's next regular biweekly *Federal Register* notice.

Sincerely,

/RA/

Alexander N. Chereskin, Project Manager
Plant Licensing Branch I-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket Nos. 50-317 and 50-318

Enclosures:
1. Amendment No. 315 to DPR-53
2. Amendment No. 293 to DPR-69
3. Safety Evaluation

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CALVERT CLIFFS UNIT 1

AMENDMENT NO. 293 TO RENEWED FACILITY OPERATING LICENSE NO. DPR-69
CALVERT CLIFFS UNIT 2

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