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

July 23, 2015

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

SUBJECT: CALVERT CLIFFS NUCLEAR POWER PLANT, UNIT NOS. 1 AND 2 - AMENDMENT REGARDING NEW TECHNICAL SPECIFICATION 3.7.18 FOR ATMOSPHERIC DUMP VALVES (TAC NOS. MF3388 AND MF3389)

Dear Mr. Gellrich:

The Commission has issued the enclosed Amendment No. 311 to Renewed Facility Operating License No. DPR-53, and Amendment No. 289 to Renewed Facility Operating License No. DPR-69 for the Calvert Cliffs Nuclear Power Plant, Unit Nos. 1 and 2. These amendments consist of changes to the Technical Specifications (TSs) in response to your application transmitted by letter dated January 13, 2014, as supplemented by letters dated November 3, 2014, March 3, 2015, and March 27, 2015.

These amendments revise the TSs to add a new TS 3.7.18, "Atmospheric Dump Valves (ADVs)." The addition of these TSs addresses a degraded or non-conforming condition that was caused by not having TSs for the ADVs. Currently the licensee is using a section in their Technical Requirements Manual (TRM) to mimic the proposed changes. The section in the TRM will only be used prior to issuance of this amendment.

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. 311 to DPR-53
2. Amendment No. 289 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. 311
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 January 13, 2014, as supplemented by letters dated November 3, 2014, March 3, 2015, and March 27, 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. <u>Technical Specifications</u>

      The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 311, 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 within 60 days.

FOR THE NUCLEAR REGULATORY COMMISSION

Michael I. Dudek

Michael I. Dudek, Acting 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: July 23, 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. 289
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 January 13, 2014, as supplemented by letters dated November 3, 2014, March 3, 2015, and March 27, 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. 289, 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 within 60 days.

FOR THE NUCLEAR REGULATORY COMMISSION

Michael I. Dudek, Acting 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: July 23, 2015

ATTACHMENT TO LICENSE AMENDMENTS

AMENDMENT NO. 311 TO RENEWED FACILITY OPERATING LICENSE NO. DPR-53

AMENDMENT NO. 289 TO RENEWED FACILITY OPERATING LICENSE NO. DPR-69

DOCKET NOS. 50-317 AND 50-318

Replace the following page of the Renewed Facility Operating License with the attached revised page. The revised page is identified by amendment number and contains marginal lines indicating the areas of change.

Remove Page	Insert Page
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
	3.7.18-1
	3.7.18-2

(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. 311, 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. 289 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

3.7 PLANT SYSTEMS

3.7.18 Atmospheric Dump Valves (ADVs)

LCO 3.7.18 Two ADV lines shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3,
MODE 4 when steam generator is being relied upon for heat removal.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One required ADV line inoperable.	A.1 Restore ADV line to OPERABLE status.	48 hours
B. Two ADV lines inoperable.	B.1 Restore one ADV line to OPERABLE status.	1 hour
C. Required Action and associated Completion Time not met.	C.1 Be in MODE 3. <u>AND</u> C.2 Be in MODE 4 without reliance upon steam generator for heat removal.	6 hours 24 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.18.1	Verify one complete cycle of each ADV.	24 months

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SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO ADDITION OF TECHNICAL SPECIFICATION 3.7.18

ATMOSPHERIC DUMP VALVES

AMENDMENT NO. 311 TO RENEWED FACILITY OPERATING LICENSE NO. DPR-53

AMENDMENT NO. 289 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 January 13, 2014 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML14015A138), as supplemented by letters dated November 3, 2014 (ADAMS Accession No. ML14309A717), March 3, 2015 (ADAMS Accession No. ML15065A030), and March 27, 2015 (ADAMS Accession No. ML15090A192), Exelon Generation Company, LLC (Exelon, the licensee), submitted a request for changes to the Calvert Cliffs Nuclear Power Plant, Unit Nos. 1 and 2 (CCNPP or Calvert Cliffs), Technical Specifications (TSs). The supplements dated November 3, 2014, March 3, 2015, and March 27, 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 42548).

The proposed changes would revise the TSs to add TS 3.7.18, "Atmospheric Dump Valves (ADVs)." The addition of these TSs addresses a degraded or non-conforming condition that was caused by not having TSs for the ADVs. Currently the licensee is using a section in their Technical Requirements Manual (TRM) to mimic the proposed changes. The section in the TRM will only be used prior to issuance of this amendment.

Enclosure

2.0 REGULATORY EVALUATION

The proposed License Amendment Request (LAR) dated January 13, 2014, adds the Limiting Conditions for Operation (LCO) for TS 3.7.18 as follows:

> Two ADV lines shall be OPERABLE.

The "ACTIONS" table associated with LCO 3.7.18 is as follows:

1. For one ADV line inoperable, a Completion Time (CT) of 48 hours is established to restore the ADV line to operable status.

2. For two ADV lines inoperable, a CT of 1 hour is established to restore at least one ADV line to operable status.

3. If those Required Actions, and CT are not met, the Unit would have to be in Mode 3 in 6 hours and Mode 4 on shutdown cooling in 24 hours.

The licensee also stated in its letter dated January 13, 2014, that:

> A Surveillance Requirement (SR) is also proposed to require one complete cycle of each ADV once every 24 months (refueling interval).

The licensee proposed the addition of TS 3.7.18 to ensure that the number of ADV lines maintained in an operable condition will allow each unit to be safely shut down after a design basis event concurrent with a loss of offsite power (LOOP) and, as claimed by the licensee, assuming a single failure.

2.1 Component Description

Calvert Cliffs Updated Safety Analysis Report (UFSAR), Section 10.1, "Main Steam System [MSS]," describes the MSS as being designed to transfer steam from the steam generators (SGs) to the turbine throttle stop valves, the reheaters, and the turbine-driven pumps. The MSS also controls SG pressure by means of the steam bypass, dump, safety valves (high pressure), and the main steam isolation valves (MSIVs) (low pressure).

Calvert Cliffs UFSAR Section 10.1.2.2, "Atmospheric Steam Dumps and Bypass System," provides a brief description of the ADVs and is supplemented by the additional design information provided by the licensee in this amendment.

The licensee stated in its letter dated January 13, 2014, that:

> The steam dump and bypass system is used to rapidly remove Reactor Coolant System (RCS) stored energy and to limit secondary steam pressure following a turbine and reactor trip. The steam dump system consists of two ADVs (one per steam generator) which exhaust to the atmosphere.

> The steam dump system is safety-related.
>
> Two normally shut ADVs are connected to the main steam headers between the containment penetrations and the MSSVs [main steam safety valves]. When opened, the ADVs exhaust part of the secondary steam flow to the atmosphere through separate vent enclosures which extend from the 45-foot level up through the roof of the Auxiliary Building.
>
> There is one ADV per main steam line (MSL) and one MSL per steam generator (SG). The ADVs are..... air-operated (air to open), 5-inch globe valves that are made of carbon steel. They have a steam flow capacity of 281,750 lbm/hr [pounds/hour]. Combined, the two valves are capable of passing 5% of the total secondary steam flow. This rating enables them to remove reactor decay heat during plant cooldown or heatup.
>
> The valves are designed to fail in the shut position. Each valve is equipped with a manual override (hand wheel) to allow it to be locally manually operated as required. The ADVs can be isolated using a manually operated isolation valve that is installed upstream of each ADV inlet.
>
> The ADV controls provide automatic or operator control of the ADVs during normal and emergency plant operation. During normal plant operation, the ADVs remain shut until the main turbine trips.

There are three modes of operation for the ADVs. The licensee also stated in its January 13, 2014, letter that:

> They can be automatically operated as part of the Reactor Regulating system. They can also be manually controlled (via Hand Indicating Controllers) from the Control Room [CR] or the alternate shutdown panel. They can also be manually controlled locally (via a hand wheel) at the ADV enclosures in the Auxiliary Building. The system controls are arranged for either automatic operation or remote manual control. Normal operation is through the Reactor Regulating system.
>
> Usually, the ADVs are positioned by the Reactor Regulating system, using the reactor coolant average temperature error signal when the turbine is tripped. In the event of a turbine trip above a preset power level, a quick-opening signal is provided to fully open both the ADVs and the turbine bypass valves. When T_{avg} is reduced to less than a predetermined temperature, the ADVs are modulated as a function of T_{avg}, and the turbine bypass valves are modulated as a function of the main steam header pressure. The total respective capacities of the ADVs and turbine bypass valves are 5% and 40% of full reactor power steam flow.
>
> If manual operation is needed, the ADV can be manually controlled using Hand Indicating Controllers (HICs) in the CR. If operation outside the CR is needed, the ADVs can also be manually controlled from the auxiliary shutdown panel. The ADV hand controllers each control one ADV and are equipped with a variable valve position control with a 0 to 100% output meter. Control of the

> ADVs is shifted from the CR to the auxiliary shutdown panel using four control transfer switches located in the 45-foot Switchgear Room. The quick-opening feature is disabled.
>
> If local manual operation is required, the ADVs can be locally operated using a hand wheel attached to the ADV. The hand wheel is external to the ADV enclosure in the Auxiliary Building. The area is accessible following a turbine and reactor trip or an accident. Intermediate positioning of the ADV can also be performed using the hand wheel.
>
> The ADVs controls receive electrical power from emergency diesel generator-backed, engineered safety feature, 125 VDC [Ventilation Duct Chase] unit control panels. When electrical power is unavailable, the quick-opening feature is disabled. The ADVs may still be automatically or manually controlled from the CR. Loss of control voltage also actuates an alarm in the CR. Local manual operation of the ADVs does not require electrical power or air to function as designed.

Under accident conditions, the licensee stated:

> The ADV has two functions to be considered. One function of the ADV is to cool the RCS to shutdown cooling (SDC) conditions following an accident by passing steam from the main steam system to the atmosphere at a rate equivalent to 5% full reactor power steam flow. This can be accomplished with local manual (hand wheel) operation. The other specified function of the ADV is to close when needed to minimize a radioactive release via its associated steam generator (SG) due to a leak from the RCS to the secondary system. In the bounding case from the accident analyses in the UFSAR, closure of an ADV is required within two hours following reactor trip for a SGTR [Steam Generator Tube Rupture]. This can be accomplished with local manual (hand wheel) operation.

Calvert Cliffs UFSAR, Section 10.1, "Main Steam System," describes the MSS as being designed:

> ...to transfer steam from the steam generators to the turbine throttle stop valves, the reheaters, and the turbine-driven pumps. The MSS also controls SG pressure by means of steam bypass, dump, or safety valves (high pressure) and Main Steam Isolation Valves (MSIVs) (low pressure).

The ADVs, which are described in Calvert Cliffs UFSAR Section 10.1.2.2, "Atmospheric Steam Dump and Bypass System," as part of the atmospheric steam dump system within the Main Steam Supply System (MSSS), are provided to allow cooldown of the SGs when the MSIVs are closed, or when the main condenser is not available as a heat sink. Each atmospheric dump valve is sized to hold the plant at hot standby while dissipating core decay heat (up to 5% of full reactor power steam flow) or to allow a flow of sufficient steam to maintain a controlled reactor cooldown rate. Along with one ADV per MSL, the MSSS includes main steam piping from the SG nozzles to the main turbine stop valves, one main steam isolation valve per MSL, and eight main steam safety valves per MSL.

2.2 Applicable Regulatory Requirements

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 Title 10 of the *Code of Federal Regulations* (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 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."

The licensee has proposed a new TS based on the requirements of set forth in 10 CFR, Section 50.36, "Technical Specifications." ADVs lines are being added to the TS since they meet Criterion 3 of 10 CFR 50.36(c)(2)(ii) which states that:

> A structure, system, or component that is part of the primary success path and which functions or actuates to mitigate a DBA [design basis accident] or transient that either assumes the failure or presents a challenge to the integrity of a fission product barrier.

The ADV lines are utilized in the accident analysis and in the emergency procedures. For example, the UFSAR accident analysis in Section 14.15 describes that closure of the ADV is required within two hours following a reactor trip for a SGTR.

The requirement for LCOs in 10 CFR 50.36(c)(2) states as follows:

> Limiting conditions for operation are the lowest functional capability or performance levels of equipment required for safe operation of the facility. When a limiting condition for operation of a nuclear reactor is not met, the licensee shall shut down the reactor or follow any remedial action permitted by the technical specifications until the condition can be met.

The regulation at 10 CFR 50.36(a)(1) states:

> A summary statement of the bases or reasons for such specifications ... shall also be included in the application, but shall not become part of the technical specifications.

The regulations at 10 CFR 50.36(c) state that the TSs include items in five categories. These categories include (1) safety limits, limiting safety system settings, and limiting control setting, (2) LCOs, (3) SRs, (4) design features, and (5) administrative controls.

The regulations in 10 CFR 50.36(c)(2)(ii) set forth four criteria to be used in determining whether an LCO is required to be included in TSs. These criteria are:

(A) Criterion 1. Installed instrumentation that is used to detect, and indicate in the control room, a significant abnormal degradation of the reactor coolant pressure boundary.

(B) Criterion 2. A process variable, design feature, or operating restriction that is an initial condition of a design-basis accident or transient analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.

(C) Criterion 3. 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.

(D) Criterion 4. A structure, system, or component which operating experience or probabilistic risk assessment has shown to be significant to public health and safety.

The ADVs are part of the primary success path for cooldown of the unit following a SGTR. In a SGTR, the fission product barrier (the RCS) is assumed to have failed. Therefore, the ADVs meet 10 CFR 50.36(c)(2)(ii) Criterion 3, and should be included in the TSs. The proposed TS is based on NUREG-1432, "Standard Technical Specifications Combustion Engineering Plants," dated April 2012 and is modified based on plant specific design features.

The license has considered this lack of TS to be a degraded or non-conforming condition, and has imposed appropriate administrative controls. A TRM section was created for the ADVs which mimics this proposed TS change. This TRM will remain in place until this licensing amendment request is dispositioned.

In the LAR letter dated January, 13, 2014, the licensee stated that the safety analyses descriptions in the Calvert Cliffs UFSAR were reviewed, and found that the ADVs are not credited in any of the core response portions of the analyses because assuming that the ADVs operate would result in a less severe core response. However, the use of the ADVs is described in several accident analyses as it relates to control of radiological dose (see Table 1 of the LAR). The licensee identified 10 CFR, Part 50, Appendix A, GDCs 34 and 60 as applicable.

GDC 34, "Residual Heat Removal," states as follows:

> A system to remove residual heat shall be provided. The system safety function shall be to transfer fission product decay heat and other residual heat from the reactor core at a rate such that specified acceptable fuel design limits and the design conditions of the reactor coolant pressure boundary are not exceeded.
>
> Suitable redundancy in components and features, and suitable interconnections, leak detection, and isolation capabilities shall be provided to assure that for onsite electric power system operation (assuming offsite power is not available) and for offsite electric power system operation (assuming onsite power is not

available) the system safety function can be accomplished, assuming a single failure.

GDC 60 "Control of Releases of Radioactive Materials to the Environment," states as follows:

> The nuclear power unit design shall include means to control suitably the release of radioactive materials in gaseous and liquid effluents and to handle radioactive solid wastes produced during normal reactor operation, including anticipated operational occurrences. Sufficient holdup capacity shall be provided for retention of gaseous and liquid effluents containing radioactive materials, particularly where unfavorable site environmental conditions can be expected to impose unusual operational limitations upon the release of such effluents to the environment.

In consideration of the above GDCs, the ADVs play a role in one design basis event, namely a SGTR as the limiting case for radiological releases due to ADV operation. The specific role of the ADV in this event is discussed and evaluated further in the Technical Evaluation.

Based on the above regulatory evaluation, the NRC staff reviewed the licensee's proposed new LCO, associated SRs, and supporting analyses to determine whether the proposed revision to the TS will meet the requirements of 10 CFR 50.36 for the inclusion of the ADV TS. The results of the NRC staff's review are presented in the next section.

3.0 TECHNICAL EVALUATION

3.1 Background

The two units at Calvert Cliffs are very similar in design. Each unit has two SGs and each SG has one MSL. Each MSL has one ADV. The licensee's design basis credits the SG ADVs as the primary safety related means to remove decay heat from the RCS. The ADVs are the safety related credited equipment for mitigating accidents and transients. The safety function of the ADVs is to provide means to release energy from the SGs to the atmosphere. The ADV lines allow the SGs to perform their safety function as a heat sink for removing the heat from the RCS. The ADVs become essential when the normal method of removing heat from the SGs by steaming to the condenser is lost when off-site power is lost.

The licensee is requesting a change to TSs by the addition of a new TS 3.7.18, "Atmospheric Dump Valves."

As stated in the LAR dated January 13, 2014, that the proposed new TS 3.7.18 states:

> ...that two ADV lines shall be operable in Modes 1, 2, 3, and 4 (when the steam generator is being relied upon for heat removal). The proposed Actions address Conditions where one ADV line is inoperable and when two ADV lines are inoperable. For one ADV line inoperable, a Completion Time of 48 hours is established to restore the ADV line to operable status. For two ADV lines inoperable, a Completion Time of 1 hour is established to restore at least one ADV line to operable status. If those Required Actions and Completion Times are not met, the unit would have to be in Mode 3 in 6 hours and Mode 4 on

shutdown cooling in 24 hours. A Surveillance Requirement (SR) is also proposed to require one complete cycle of each ADV once every 24 months (refueling interval).

Also as stated in the LAR dated January 13, 2014, the proposed TS Bases 3.7.18 states:

> ...that the ADVs are provided with upstream isolation valves to permit their being tested at power, if desired. The ADVs are equipped with manual hand wheels to open and close them. Pneumatic controllers are used to operate the ADVs as the preferred method, but are not relied upon during an accident.
>
> The ADVs are considered OPERABLE when the manual control is available for local manual operation.

Specifically, the operation of the ADVs is described in USFAR Section 14.15, "Steam Generator Tube Rupture Event." Operator actions assumed in this analysis are stated by the licensee to be consistent with the Calvert Cliffs Emergency Operating Procedures (EOPs). Some of the major post-trip EOP analysis assumptions regarding operator actions related to the ADVS are:

1. ...15 minutes after reactor trip, the operator takes manual control of the ADV on the affected SG to prevent further cycling of the MSSVs.

3. The operator stabilizes the plant...The length of the stabilization period is assumed to be no more than 10 minutes from the time that the operator takes manual control of the ADVs...the operator initiates action to unisolate the ADV of the intact SG, which is assumed to be isolated at this time. The actions may take up to 1 hour after taking control.

5. [The operator] Isolate[s] the affected SG...The analysis assumes no opening of the ADV of the affected SG after 2 hours. However, the ADV of the affected SG may be opened 24 hours into the accident to hasten shutdown.

6. Following the isolation of the affected SG, the operator cools down the plant using the ADV on the intact SG...

As stated in the LAR dated January 13, 2014:

> The ADV controls provide automatic or operator control of the ADVs during normal and emergency plant operation. During normal plant operation, the ADVs remain shut until the main turbine trips. There are three modes of operation for the ADVs which include automatic operation as part of the Reactor Regulating system, manual control (via Hand Indicating Controllers) from the Control Room or the alternate shutdown panel, and manual control locally (via a hand wheel) at the ADV enclosures in the Auxiliary Building. The system controls are arranged for either automatic operation or remote manual control. Normal operation is through the Reactor Regulating system.

3.2 Staff Evaluation of the Design Basis Event

The licensee's submittal letter, dated January 13, 2014, stated that the SGTR event requires ADVs for mitigation.

The NRC staff's review of the design basis events will look to establish that:

1. The proposed LCO is consistent with the existing UFSAR safety analyses.
2. The plant has adequate defense in depth to mitigate the affected design basis events, should one of the Calvert Cliffs nuclear power plants enter the proposed conditions associated with the LCO.
3. The ADV design mitigates accidents with a loss of offsite power.
4. The ADV testing and Failure Modes and Effects (FME) analysis is acceptable.
5. The ADV TS CT and block valve TS SRs are acceptable.
6. The closing of the ADVs with manual controls is acceptable.

The proposed LCO will require both ADVs to be OPERABLE, but will allow up to one ADV on either SG to be inoperable for up to 4 hours. The proposed LCO will also allow both ADVs to be inoperable for no more than 1-hour.

The existing UFSAR safety analyses, including the computer codes which form the basis of the SGTR analysis, were not reviewed in substantial detail by the NRC staff as part of this LAR review. However, the licensee has provided supplemental analyses to justify the proposed conditions, required actions, and CTs. These supplemental analyses are reviewed in detail.

In the LAR dated January 13, 2014, the licensee stated that:

> The SGTR analysis is the limiting case for radiological releases due to ADV operation. The ADV line on the affected SG is assumed to open upon turbine and reactor trip and a loss of offsite power. The condenser is not available and the turbine bypass valves remain closed. The MSSVs also mitigate the initial pressure increase and help to cool the RCS within the limits of their setpoints. Within one hour after initially taking control, the ADV on the unaffected SG is opened (if not previously opened) to begin an RCS cooldown and minimize any additional MSSV operation. The one hour time allows for local manual operation of the ADV using the hand wheel at the ADV enclosure in the Auxiliary Building, if automatic or remote manual operation does not work. If the ADV line is isolated with the installed isolation valve, the isolation valve is also manually opened. The isolation valve is located in the main steam isolation valve room, just below the ADV enclosure. At two hours after turbine and reactor trip, the ADV on the affected SG is closed and the radiological releases associated with it are terminated.

3.2.1 Steam Generator Tube Rupture

The licensee's January 13, 2014, LAR stated the following with respect to the UFSAR accident analysis:

> To address the scope of the proposed TS, the UFSAR accident analyses descriptions were reviewed to determine the extent to which the ADVs are credited in the accident analyses. The review determined that the ADVs are not credited in any of the core response portions of the analyses, because assuming that the ADVs operate would result in a less severe core response. However, ADV use is described in several accident analyses as it relates to control of radiological dose.

The licensee determined that the SGTR analysis is the limiting case for radiological releases due to ADV operation. The SGTR accident analysis is presented in Section 14.15 of the Calvert Cliffs UFSAR. The staff also determined that the SGTR accident analysis of UFSAR Section 14.15 originated as part of a LAR for a revision to the alternative source term (AST) methodology described in Regulatory Guide 1.183 (dated November 3, 2005, ADAMS Accession No. ML053200300).

The behavior of the primary and secondary systems during and after a double-ended tube break SGTR event was discussed in Enclosure 4 of the AST LAR package (i.e., "CA06453 SGTR Radiological Consequences Design Basis Calculation Using AST" (ADAMS Accession No. ML053210454)). LAR Enclosure 4 and UFSAR Section 14.15 cite a Westinghouse calculation for the re-analyzed SGTR event as Reference 5 in the AST LAR and as Reference 4 of UFSAR Section 14.15. The Westinghouse calculation CN-TAS-05-13, Revision 000, "Calvert Cliffs Units 1 & 2 Steam Generator Tube Rupture Event," is dated August 12, 2005.

The NRC staff determined that additional information was necessary to determine the appropriate TS conditions, TS actions, and TS SR. The staff issued a request for additional information (RAI) letter dated September 17, 2014 (ADAMS Accession No. ML14237A069), that questioned the Calvert Cliffs plant design of one ADV per MSL, which does not meet single failure consideration. The staff questioned how the ADVs are utilized in accident mitigation, and other secondary-side steam dump pathways for plant cooldown as described in the UFSAR. The staff also specifically requested that the licensee provide additional information to justify its proposed LCO, with respect to the SGTR event, by providing the calculation file (CN-TAS-05-13) referenced in the UFSAR as containing the technical evaluation of a SGTR event with LOOP.

The licensee responded to this RAI request in a letter dated November 3, 2014 (ADAMS Accession No. ML14309A717), and stated that the two ADVs have a combined capacity of passing 5 percent of the total secondary steam flow. This rating enables them to remove reactor decay heat during plant cooldown or heatup. The original design of the atmospheric dump system did not include two valves per SG. The original SGTR analyses assumed that the steam from the faulted SG was directed to the condenser through the turbine bypass system. The dose analysis at the time assumed all radioactivity was released through the condenser air removal system and not through the ADVs. As the SGTR analysis was updated over time, and when it was updated to take advantage of the alternate source term, the exclusive use of the

ADVs for natural circulation cooldown was assumed to maximize offsite dose. The physical plant design did not change, and as stated by the licensee, the ADVs do not meet single failure considerations.

The licensee's RAI response also included the requested calculation file (CN-TAS-05-13), which is proprietary to Westinghouse but forms the basis for the discussion of the SGTR event in the UFSAR. Thus, the NRC staff reviewed CN-TAS-05-13 to confirm that statements made in the LAR are reasonable and that the assumptions, information, and results in the UFSAR for the operation of the ADVs were consistent with the proprietary calculation.

From UFSAR Section 14.15, the major post-trip EOP analysis assumptions regarding operator actions are:

1. <u>Operate the ADV on the affected SG</u>: 15 minutes after reactor trip, the operator takes manual control of the ADV on the affected SG to prevent further cycling of the MSSVs.

2. <u>Take manual control of the Auxiliary Feedwater System (AFW) to the SGs</u>: Two minutes after opening the affected SG ADV, the operator takes manual control of the AFW flow to each SG, with flow initially delivered to both SGs.

3. <u>Stabilize the plant and maintain cold leg temperature</u>: The operator quickly diagnoses the event and stabilizes the RCS to a temperature which precludes a challenge to the MSSVs using the SG ADVs and AFW. The length of the stabilization period is assumed to be no more than 10 minutes from the time that the operator takes manual control of the ADVs. As a result of this diagnosis, the operator initiates action to unisolate the ADV of the intact SG, which is assumed to be isolated at this time. The actions may take up to 1 hour after taking control.

4. <u>Cool the RCS before isolating the Affected SG</u>: After the stabilization period, the operator begins to cool the RCS at a rate of up to 100°F per hr to maximum steam releases.

5. <u>Isolate the Affected SG</u>: The operator isolates the affected SG when T_{HOT} is less than 515°F (including uncertainties). The analysis assumes no opening of the ADV or MSSVs of the affected SG after 2 hours. However, the ADV of the affected SG may be opened 24 hours into the accident to hasten shutdown.

6. <u>Plant cooldown after isolation of the affected SG</u>: Following the isolation of the affected SG, the operator cools down the plant using the ADV on the intact SG at a maximum of 35°F/hr to maximize steam releases.

7. <u>Maintain SG pressure and level</u>: The pressure and level of the affected SG will initially be controlled by steaming to atmosphere for up to 2 hours. In addition, the RCS will be aggressively cooled down to achieve backflow from the affected SG as early in the event as possible.

8. Maintain subcooling margin during the event: A target subcooling margin of 50°F is maintained by the operator. This value consists of 25°F required by the EOPs and 25°F of core exit thermocouple uncertainty.

9. Maintain pressurizer level: The pressurizer level is maintained by controlling safety injection flow. In addition, the RCS is aggressively cooled down to achieve backflow from the affected SG as early in the event as possible.

10. Pressurizer control actions and control systems: The operator uses the High-Pressure Safety Injection system and the pressurizer vent (or auxiliary spray) to control RCS inventory and subcooling.

As described in UFSAR Section 14.15, the operator would continue to perform the natural circulation cooldown per the EOPs until the SDC conditions (T_{ave} less than or equal to 300°F and RCS pressure to be less than 270 psig (page 12 of 114 of AST LAR Enclosure 4, ADAMS Accession No. ML053210454)) are attained, and releases from the steam generators have been terminated.

The staff confirmed that the above assumptions that were part of the AST LAR, and the current UFSAR Section 14.15 assumptions were also applied in the SGTR accident analysis of CN-TAS-05-13.

The analysis was performed with the goal that it would be used to produce CR doses. This was done in the AST LAR which was reviewed, and accepted by the NRC staff as shown in the issued amendments and enclosed SE (Amendment #281 dated August 29, 2007, ADAMS Accession No. ML072130521).

The supplemental information contained in the proprietary calculation file included tables and transient plots of the SGTR event. The NRC staff reviewed the SGTR calculations and determined that the assumptions were supported by the analysis. The operator begins to cooldown the plant to limit MSSV cycling (note: the MSSVs would limit the SG shell side pressure). The SG pressure, SG level, pressurizer level, and primary pressure control are maintained by manual operation actions. The plant cooldown rate and subcooled margin can be maintained but may not be at the target cooldown rate values when relying on just the unaffected ADV as the controlling component. As discussed in the AST LAR SE Section 3.2.4.2, "[SGTR] Transport Methodology and Assumptions" (ML072130521), the operator has two options that rely on safety-related equipment which would be appropriate for mitigating the design basis SGTR accident:

- The operator continues the cooldown via the ADV of the unaffected SG until SDC entry conditions are reached. It will take approximately 14 days for the decay heat generation to decline to a level that can be removed via a single SG and ADV. Instead of assuming a 0 - 2-hour cooldown via the ADV of the affected SG, followed by a 2 - 30-day cooldown via the unaffected SG, the licensee conservatively assumes a 0 - 30-day cooldown via the ADV of the unaffected SG to model this mode.

- The operators can re-open the ADV of the affected SG for up to 8 hours after an initial cooldown of 24 hours post-accident. The licensee models this by assuming an initial 0 - 2-hour cooldown via the ADV of the affected SG, followed by a 2 - 24-hour cooldown via the ADV of the unaffected SG, then a 24 - 32-hour cooldown via the ADV of the affected SG.

Thus, the Westinghouse calculation contained in the AST LAR and in the supplemental information demonstrates the importance for the availability of both ADVs to mitigate the design basis SGTR accident in the least amount of time. Based on these considerations, the NRC staff finds that the proposed TS conditions and required actions are acceptable.

3.2.2 Availability of Other Means for Controlling Natural Circulation Cooldowns

During the review LAR, the NRC staff found statements in the LAR that were not consistent with Revision 43 of the UFSAR. In particular, on Page 5 of the LAR dated January 13, 2014, the licensee stated that for the SGTR event under UFSAR Section 14.15:

> The ADV on the affected SG is assumed to open upon turbine and reactor trip and a loss of offsite power [LOOP].

However, UFSAR Section 14.15 does not provide a LOOP as an assumption for the presented analysis. The UFSAR states that for the SGTR event analysis, "no credit was taken for the operation of the steam bypass valves to the condenser," it is later stated that other means are available for cooldown, if the ADVs are unavailable.

Therefore, the NRC staff requested that the licensee provide documentation (analysis and procedures) as to what other means are available for cooldown if the ADVs are unavailable given the event assumptions of a LOOP and no credit being taken for the operation of the steam bypass valves to the condenser (RAI letter dated September 17, 2014, ADAMS Accession No, ML14237A069). In the RAI response dated November 3, 2014, the licensee revised UFSAR Section 14.15 to:

> ...reflect the current understanding that a loss of offsite power is integral to the event, since it will drive all of the steam from the faulted SG to the atmosphere through the associated ADV. This will maximize the dose for the dose analysis.
>
> In reviewing UFSAR Section 14.15, it was determined that the statements related to other means of plant cooldown were not adequately supported by evaluation or analysis and they have been removed. This determination provided the basis for the need to request this license amendment request, since now the ADVs were the primary success path for plant cooldown as described in the emergency procedures for a SGTR.

The NRC staff reviewed the revised UFSAR Section 14.15 pages in the RAI response dated November 3, 2014 (ADAMS Accession No. ML14309A717), and found them acceptable.

3.2.3 Staff Summary of SGTR Accident Analysis Review

The NRC staff confirmed the applicant's statements in the LAR by referring to the licensee

prepared calculation package CN-TAS-05-13. Specifically the staff confirmed that the assumptions used in the analysis are consistent with the licensing basis, the approach used, and that the statements made in the LAR are reasonable.

3.2.4 ADV Design and Accident Mitigation with Loss of Offsite Power

During the review of the LAR, the NRC staff determined that additional information was necessary to complete the review. The staff questioned how the ADV design meets single failure consideration, and how the ADVs are utilized in accident mitigation because Calvert Cliffs is designed to have one ADV per MSL. The staff sent the licensee a RAI by letter dated September 17, 2014 (ADAMS Accession No, ML14237A069).

The licensee responded to this RAI request by letter dated November 3, 2014 (ADAMS Accession No. ML14309A717), and stated that:

> Combined, the two valves are capable of passing 5% of the total secondary steam flow. This rating enables them to remove reactor decay heat during plant cooldown or heatup.
>
> The ADVs do not meet single failure considerations. The original design of the atmospheric dump system did not include two valves per SG. The original steam generator tube rupture (SGTR) analyses assumed that the steam from the faulted SG was directed to the condenser through the turbine bypass system. The dose analysis at the time assumed all radioactivity was released through the condenser air removal system and not through the ADVs.
>
> As the SGTR analysis was updated over time, and especially when it was updated to take advantage of the Alternate Source Term, the use of the ADVs was assumed to maximize offsite dose. The physical plant design did not change.
>
> The ADV stem travels in the downward direction, into the steam space to open the valve. The valve is equipped with a pilot to assist in initial opening.
>
> The ADV is an air-to-open, fail closed, valve and air is applied to the top of the valve diaphragm in order to lower the valve stem, and open the valve. Air acts against the large spring compressing it, in order to open the valve. With no air on the diaphragm, the spring forces the valve shut.
>
> A manual operator is provided with the valve. The operator forces a rod into the valve that will act to compress the spring. The manual operator does not function to close the valve. The manual operator can be inserted to open the valve, or removed to allow the valve to close.
>
> Remote operation is the preferred method of operation; however, it is not credited in the accident analyses to mitigate the consequences of the event.
>
> Local (manual handwheel) operation of the valve is credited to mitigate the consequences of some events. As described in the Updated Final Safety

> Analysis Report (UFSAR), Section 14.15, Steam Generator Tube Rupture, this event is the most limiting event for ADV operation. In this event, the ADV does not have to be operated for up to two hours following initiation of the event. Again, remote operation is preferred, but not required.

Related to accident mitigation, the licensee also stated in its RAI response of November 3, 2014:

> Local (manual handwheel) operation of the valve is credited to mitigate the consequences of some events. For example UFSAR, Section 14.15, "Steam Generator Tube Rupture," is the most limiting event for ADV operation. In this event, the ADV does not have to be operated for up to two hours following initiation of the event. Remote operation is preferred, but not required. The ADVs need to be opened and closed manually to mitigate the consequences of some events.
>
> In order to take manual control of the ADV, operators shifts ADV control from the Control Room to the safe shutdown panel. From the safe shutdown panel, the hand controller of the ADV to be manually operated is taken to 0%, sending the ADV a full shut signal. The ADV is then manually operated as necessary.

The NRC staff reviewed the RAI response and found the response acceptable. The description and clarification of how the ADV is operated manually was described adequately to understand how the ADV is operated during a LOOP event.

3.2.5 <u>ADV Testing and FME Analysis</u>

During the review of the LAR, the NRC staff determined that additional information was necessary to complete the review. The staff issued a RAI letter dated September 17, 2014 (ADAMS Accession No, ML14237A069), to gain further understanding of the ADV testing and FME based on the Calvert Cliffs design of having one ADV per MSL.

The licensee responded to this RAI request by letter dated November 3, 2014 (ADAMS Accession No. ML14309A717), and stated that:

> A failure modes and affects analysis was performed to document conditions that could cause an ADV to either spuriously open, or not fully shut. The failures were divided into two sub-groups, mechanical, and controls.
>
> The mechanical failure modes considered are: FME in valve seat, mechanical binding, spring failure, and failure of the handwheel to retract.
>
> The mechanical failure modes are considered to be controlled by preventative maintenance practices.
>
> The control failure modes considered are: spurious air from the quick open solenoid valve, failure of the positioner to bleed off control air, and spurious signal from an I/P converter.

The NRC staff reviewed the RAI response and found the response incomplete. The licensee provided only the mechanical failure modes (which are considered to be controlled by preventative maintenance practices) and the control failure modes. However, the TS function of the ADVs is to be able to manually control a reactor cooldown post SGTR. There was no information provided that confirmed testing included the verification of ADV full-cycle with local manual operation and handwheel.

The NRC staff requested that the licensee justify why there is not a TS surveillance for the ADV line by local manual operation. For example, another plant TS SR states to verify each atmospheric dump line by local manual operation. The proposed Calvert Cliff's TS SR 3.7.18.1 states to verify one complete cycle of each ADV. The actual accident function of the ADV is with the local manual operation via the ADV hand wheel extension (reach rod) and chain operator. In addition, the staff requested the licensee provide a discussion of how the ADV manual operator with chain operator (and associated valve extension components) is designed to withstand a design bases event including a seismic event. By RAI letter dated February 2, 2015 (ADAMS Accession No. ML15027A144), the staff asked the licensee to describe the seismic classification of the ADV valve operator with chain operator including such components as the reach rod linkage, gear box, shaft, and associated linkage and gear box supports.

The licensee responded to this RAI request in two letters dated March 3, 2015 (ADAMS Accession No. ML15065A030), and March 27, 2015 (ADAMS Accession No. ML15090A192). The licensee stated that:

> The [ADV] handwheel assembly consists of a gear box, a handwheel with chains, shafts, supports, and associated linkage. The assembly is located externally, on the top of the ADV enclosure. A short portion of the linkage projects down through the top cover plate of the enclosure. The ADV enclosure is located in the Auxiliary Building and is therefore protected from external events, such as tornados and hurricanes. It is not located in an area subject to accident environmental condition extremes of temperature or radiation for the accidents of concern. It remains accessible to operators post-accident.
>
> The handwheel assembly is installed to satisfy AQ [Augmented Quality]-II/I seismic requirements. The handwheel assemblies are of rugged construction and mounted on plate steel stanchions, which are welded to the ADV enclosures.

The licensee also stated that the seismic capability of the ADV enclosures, manual operator assembly, and their potential seismic interactions (seismic II/I interactions, i.e., interactions of structures, piping, or equipment with nearby safe-shutdown equipment) were evaluated in the Unresolved Safety Issue (USI) A-46 Program implemented in the 1990's and demonstrated to be adequate without potential adverse seismic II/I interactions. In addition, a recent assessment of the new replacement chain wheel, and in-line gear assemblies subjected to a design basis seismic demand, affirmed that these assemblies would not adversely impact the structural integrity of their supports, the ADV enclosures, or the nearby Seismic Category I equipment because of the structural robustness of the welded supports on top of the ADV enclosures.

The licensee further stated in its RAI response of March 27, 2015, that:

> By observation [by the licensee] of the structural robustness of the welded

> supports it was determined that the new replacement chain wheel and in-line gear assemblies, subjected to design basis seismic demand accelerations, will not adversely impact the structural integrity of their supports, the ADV enclosures, or nearby Category I equipment. The support brackets are sufficiently stiffened and welded to preclude them from developing significant bending or shear stresses. The chain wheel and in-line gear assembly supports are acceptable.
>
> The ADVs and the manual operator have a maintenance rule function and are included in the Maintenance Rule Program.

In addition, the licensee stated in its RAI response of March 3, 2015, that:

> The cycling of the ADV, currently performed each refueling outage, is done manually using the handwheel. The intent of the Technical Specification Surveillance Requirement proposed was to continue to perform the same local manual test to ensure that the safety function of the valve is tested.

The NRC staff verified that the proposed TS Bases have been updated to clarify this information.

Based on the information provided by the licensee, the ADV manual operator assembly is not classified as Seismic Category I system but is installed to satisfy AQ II/I seismic requirements. As such, the assembly must be evaluated for its potential interaction with Seismic Category I systems, structures, and components (commonly referred to as the seismic II/I interaction), to ensure that its failure under the SSE will not adversely impact the integrity of Seismic Category I SSCs. The licensee's evaluation in the USI A-46 Program demonstrated there would be no adverse seismic II/I interactions involving the ADV assemblies, and the NRC staff's Safety Evaluation Report for USI A-46 Program implementation (1999) indicated no such potential interactions. Further, a licensee's recent assessment of the new replacement chain wheel and in-line gear assemblies affirmed that there would be no potential interaction of these assemblies with adjacent Seismic Category I SSCs when subjected to the SSE ground motion due to the rugged construction and robustness of the welded supports on top of the ADV enclosures.

The NRC staff found the licensee's response to the RAI acceptable since it clarified design functions (seismic II/I) of the ADV manual operator and that testing of the ADV's includes the reach rod assembly and handwheel. The applicant provided information that the ADV manual operator with associated extension components are designed to satisfy AQ II/I seismic requirements in such a way that it will not impair the integrity of nearby Seismic Category I SSCs during a design basis earthquake. The staff noted, in Section 5A.2 "Classes of Structures, Systems, and Equipment" of the UFSAR, Revision 34, that the Auxiliary Building below Elevation 69 feet is classified as Seismic Category I structure and that the ADV enclosure is classified as Seismic Category I equipment. The staff also noted that the ADV enclosure and handwheel assembly are located below Elevation 69 feet in the Auxiliary Building. Since the ADV enclosure and handwheel assembly are located in a Seismic Category I structure, they will be protected from external events such as tornados and hurricanes. Further, because the ADV enclosure itself is Seismic Category I equipment, it is capable of withstanding the effects of the Safe Shutdown Earthquake (SSE). In addition, the staff determined changes to TS SR 3.7.18.1 are acceptable since it includes the manual handwheel assembly.

3.2.6 Atmospheric Dump Valve TS CT and Block Valve TS SRs

During review of the LAR, the NRC staff determined that additional information was necessary to complete the review. The staff issued a RAI letter dated September 17, 2014 (ADAMS Accession No. ML14237A069) to gain an understanding of the ADV TS CT and why the ADV block valve is not included in a TS SR.

The licensee responded to this RAI request in a letter dated November 3, 2014 (ADAMS Accession No. ML14309A717), and stated the ADV CT of 48 hours was chosen to be consistent with the CT of 48 hours at a sister CE plant that has only one ADV per SG. For the ADV block valve, the valve is not assumed to be operated in response to any accident scenario. Therefore, it does not meet the requirements of 10 CFR 50.36 for inclusion in the TS. The staff reviewed the RAI response and found the response to be acceptable since the staff has previously accepted the 48 hour CT at another CE plant (one ADV being inoperable). This CT of one ADV inoperable in accordance with NUREG-1432 is 7 days and 24 hours if two ADVs are inoperable. This takes into account the redundant capability of the remaining operation ADVs. At Calvert Cliffs, with one ADV inoperable, there remains one other operable ADV for redundancy for the SGTR event.

As discussed in the licensee's RAI response, the ADV block valve is not assumed to be a functional component in the accident analysis.

3.2.7 Closing of the ADVs with Manual Controls (ADV Spontaneously Opens)

During the review of the LAR, the NRC staff determined that additional information was necessary to complete the review. The staff issued a RAI letter dated September 17, 2014 (ADAMS Accession No, ML14237A069), to gain an understanding of how the ADV is closed if the ADV spontaneously opens.

The licensee responded to this RAI request in a letter dated November 3, 2014 (ADAMS Accession No. ML14309A717), and stated that there are five possible ways the ADV could fail open:

> The ADV is an air-to-open, fail closed, valve and air is applied to the top of the valve diaphragm in order to lower the valve stem, and open the valve. Air acts against the large spring compressing it, in order to open the valve. With no air on the diaphragm, the spring forces the valve shut.
>
> In the event that the quick open solenoid valve caused the ADV to spuriously open, the airline would be manually isolated from the ADV by a handvalve from the safe shutdown panel, per station procedures.
>
> In the event that the normal current-to-pneumatic (I/P) controller caused the ADV to spuriously open, either through hand HIC failure or I/P calibration issues, the I/P would be isolated from the ADV positioner by a handvalve from the safe shutdown panel, per station procedures.
>
> In the event that the safe shutdown panel I/P controller caused the ADV to spuriously open, either through HIC failure or I/P calibration issues, the I/P would

be isolated from the ADV positioner by a handvalve from the safe shutdown panel, per station procedures.

In the event that the ADV positioner was not fully bleeding off air to the ADV diaphragm, instrument air to the ADV positioner would be isolated by the instrument air inlet valve, located outside the ADV enclosure, per station procedures.

The NRC staff reviewed the RAI response and found the responses to be acceptable since spring action in the ADV actuator would close the ADV once instrument air is removed.

3.2.8 Technical Evaluation Summary

The NRC staff found that the proposed TS changes meet the regulatory requirements of 10 CFR 50.36(c)2.i because the LCO describes the lowest functional capability or performance levels of the ADVs required for safe operation of the facility, and the "ACTIONS" tables describe the conditions when LCO is not met and require the licensee to shut down the reactor or follow any remedial action permitted by the TS until the condition can be met. The staff also found that the proposed SR meets the requirements of 50.36(c)3 because the SR will provide assurance that the necessary quality of systems and components is maintained, that facility operation will be within safety limits, and that the LCO will be met. Therefore, the NRC staff finds the proposed change to be acceptable.

Pursuant to 10 CFR 50.36(a) and 10 CFR 50.90, along with the proposed TS changes, the licensee also submitted TS Bases changes corresponding to the proposed TS changes. The NRC staff determined that TS Bases changes are consistent with the proposed TS changes and provide the purpose for each requirement in the specification consistent with the Commission's "Final Policy Statement on Technical Specifications Improvements for Nuclear Power Reactors" dated July 22, 1993 (58 FR 39132).

4.0 STATE CONSULTATION

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

5.0 ENVIRONMENTAL CONSIDERATION

The amendments change a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20, and changes surveillance requirements. 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 *Federal Register* on July 22, 2014 (79 FR 42548). 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 licensee has provided an evaluation proposing a new TS to include the ADV lines. The NRC staff performed an extensive technical review of their LAR. Based on the above, the staff found the licensee's proposed new TS 3.7.18 acceptable. In addition, the proposed addition of the ADV TS does not change compliance with any codes or standards that have been previously committed to or used. The safety analysis acceptance criteria continue to be met and the addition of the TS for the ADVs ensures that the inputs and assumptions of the SGTR are not adversely affected.

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.

Principal Contributors: D. Palmrose, L. Wheeler, S. Park

Date: July 23, 2015

DATED: July 23, 2015

AMENDMENT NO. 311 TO RENEWED FACILITY OPERATING LICENSE NO. DPR-53
CALVERT CLIFFS UNIT 1

AMENDMENT NO. 289 TO RENEWED FACILITY OPERATING LICENSE NO. DPR-69
CALVERT CLIFFS UNIT 2

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July 23, 2015

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

SUBJECT: CALVERT CLIFFS NUCLEAR POWER PLANT, UNIT NOS. 1 AND 2 - AMENDMENT REGARDING NEW TECHNICAL SPECIFICATION 3.7.18 FOR ATMOSPHERIC DUMP VALVES (TAC NOS. MF3388 AND MF3389)

Dear Mr. Gellrich:

The Commission has issued the enclosed Amendment No. 311 to Renewed Facility Operating License No. DPR-53, and Amendment No. 289 to Renewed Facility Operating License No. DPR-69 for the Calvert Cliffs Nuclear Power Plant, Unit Nos. 1 and 2. These amendments consist of changes to the Technical Specifications (TSs) in response to your application transmitted by letter dated January 13, 2014, as supplemented by letters dated November 3, 2014, March 3, 2015, and March 27, 2015.

These amendments revise the TSs to add a new TS 3.7.18, "Atmospheric Dump Valves (ADVs)." The addition of these TSs addresses a degraded or non-conforming condition that was caused by not having TSs for the ADVs. Currently the licensee is using a section in their Technical Requirements Manual (TRM) to mimic the proposed changes. The section in the TRM will only be used prior to issuance of this amendment

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. 311 To DPR-53
2. Amendment No. 289 To DPR-69
3. Safety Evaluation

cc w/encls: Distribution via Listserv

ADAMS Accession No. ML15133A144 *via dated memo

OFFICE	DORL/LPLI-1/PM	DORL/LPLI-1/LA	DSS/SRXB/BC	DSS/SBPB/BC	DE/EMCB/BC(A)
NAME	AChereskin	KGoldstein	CJackson*	GCasto	RPettis
DATE	6/15/2015	6/15/2015	1/06/2015	4/21/2015	6/18/2015
OFFICE	DSS/STSB/BC	OGC – NLO w/comments	DORL/LPLI-1/BC (A)	DORL/LPLI-1/PM	
NAME	RElliot	SUttal	MDudek	AChereskin	
DATE	6/24/2015	7/06/2015	7/22/2015	7/23/2015	

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