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

October 29, 2014

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

SUBJECT: CALVERT CLIFFS NUCLEAR POWER PLANT, UNIT NOS. 1 AND 2 – ISSUANCE OF AMENDMENT REGARDING THE ADOPTION OF TECHNICAL SPECIFICATION TASK FORCE 510, "REVISION TO STEAM GENERATOR PROGRAM INSPECTION FREQUENCIES AND TUBE SAMPLE SELECTION" (TAC NOS. MF3062 AND MF3063)

Dear Mr. Gellrich:

The Commission has issued the enclosed Amendment No. 308 to Renewed Facility Operating License No. DPR-53 and Amendment No. 286 to Renewed Facility Operating License No. DPR-69 for the Calvert Cliffs Nuclear Power Plant, Unit Nos. 1 and 2. The amendments consist of changes to the Technical Specifications (TSs) in response to your application dated October 31, 2013.

The amendments revise TS 5.5.9, "Steam Generator (SG) Program," and TS 5.6.9, "Steam Generator Tube Inspection Report," to address implementation issues associated with the inspection periods. The amendments also revised TS 3.4.18, "Steam Generator (SG) Tube Integrity," for administrative purposes.

The revisions are consistent with Commission-approved Technical Specifications Task Force Standard Technical Specifications Change Traveler 510, Revision 2, "Revision to Steam Generator Program Inspection Frequencies and Tube Sample Selection."

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,

Nadiyah S. Morgan, 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. 308 to DPR-53
2. Amendment No. 286 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. 308
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 October 31, 2013, 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. 308, 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 within 60 days.

FOR THE NUCLEAR REGULATORY COMMISSION

Benjamin G. Beasley

Benjamin G. Beasley, 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: October 29, 2014

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. 286
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 October 31, 2013, 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. 286, are hereby incorporated in 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

Benjamin G. Beasley

Benjamin G. Beasley, 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: October 29, 2014

ATTACHMENT TO LICENSE AMENDMENTS

AMENDMENT NO. 308 TO RENEWED FACILITY OPERATING LICENSE NO. DPR-53

AMENDMENT NO. 286 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
3.4.18-1	3.4.18-1
3.4.18-2	3.4.18-2
5.5-7	5.5-7
5.5-8	5.5-8
5.5-9	5.5-9
5.5-10	5.5-10
5.5-11	5.5-11
5.5-12	5.5-12
5.5-13	5.5-13
5.5-14	5.5-14
5.5-15	5.5-15
5.5-16	5.5-16
5.5-17	5.5-17
5.5-18	5.5-18
5.5-19	5.5-19
--------	5.5-20
5.6-7	5.6-7

(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) <u>Maximum Power Level</u>

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) <u>Technical Specifications</u>

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 308, 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) <u>Additional Conditions</u>

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) <u>Secondary Water Chemistry Monitoring Program</u>

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. 286 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

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.18 Steam Generator (SG) Tube Integrity

LCO 3.4.18 SG tube integrity shall be maintained.

AND

All SG tubes satisfying the tube plugging criteria shall be plugged in accordance with the Steam Generator Program.

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

ACTIONS

------------------------------------NOTES------------------------------------
Separate Condition entry is allowed for each SG tube.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more SG tubes satisfying the tube plugging criteria and not plugged in accordance with the Steam Generator Program.	A.1 Verify tube integrity of the affected tube(s) is maintained until the next refueling outage or SG tube inspection.	7 days
	AND	
	A.2 Plug the affected tube(s) in accordance with the Steam Generator Program.	Prior to entering MODE 4 following the next refueling outage or SG tube inspection

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
B. Required Action and associated Completion Time of Condition A not met. OR SG tube integrity not maintained.	B.1	Be in MODE 3.	6 hours
	AND		
	B.2	Be in MODE 5.	36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.18.1	Verify SG tube integrity in accordance with the Steam Generator Program.	In accordance with the Steam Generator Program
SR 3.4.18.2	Verify that each inspected SG tube that satisfies the tube plugging criteria is plugged in accordance with the Steam Generator Program.	Prior to entering MODE 4 following a SG tube inspection

5.5 Programs and Manuals

a. Testing frequencies applicable to the ASME Code for Operation and Maintenance of Nuclear Power Plants (ASME OM Code) and applicable Addenda as follows:

ASME OM Code and applicable Addenda terminology for inservice testing activities	Required Frequencies for performing inservice testing activities
Weekly	At least once per 7 days
Monthly	At least once per 31 days
Quarterly or every 3 months	At least once per 92 days
Semiannually or every 6 months	At least once per 184 days
Every 9 months	At least once per 276 days
Yearly or annually	At least once per 366 days
Biennially or every 2 years	At least once per 731 days

b. The provisions of SR 3.0.2 are applicable to the above required Frequencies and to other normal and accelerated Frequencies specified as 2 years or less in the Inservice Testing Program for performing inservice testing activities;

c. The provisions of SR 3.0.3 are applicable to inservice testing activities; and

d. Nothing in the ASME OM Code shall be construed to supersede the requirements of any Technical Specification.

5.5.9 Steam Generator (SG) Program

A Steam Generator Program shall be established and implemented to ensure that SG tube integrity is maintained. In addition, the Steam Generator Program shall include the following:

5.5 Programs and Manuals

a. Provisions for condition monitoring assessments. Condition monitoring assessment means an evaluation of the "as found" condition of the tubing with respect to the performance criteria for structural integrity and accident induced leakage. The "as found" condition refers to the condition of the tubing during an SG inspection outage, as determined from the inservice inspection results or by other means, prior to the plugging of tubes. Condition monitoring assessments shall be conducted during each outage during which the SG tubes are inspected or plugged to confirm that the performance criteria are being met.

b. Performance criteria for SG tube integrity. Steam generator tube integrity shall be maintained by meeting the performance criteria for tube structural integrity, accident induced leakage, and operational LEAKAGE.

   1. Structural integrity performance criterion: All in-service steam generator tubes shall retain structural integrity over the full range of normal operating conditions (including startup, operation in the power range, hot standby, and cool down), all anticipated transients included in the design specification, and design basis accidents. This includes retaining a safety factor of 3.0 against burst under normal steady-state full power operation primary to secondary pressure differential and a safety factor of 1.4 against burst applied to the design basis accident primary-to-secondary pressure differentials. Apart from the above requirements, additional loading conditions associated with the design basis accidents, or combination of accidents in accordance with the design and licensing basis, shall also be evaluated to determine if the associated loads contribute significantly to burst or collapse. In the assessment of tube integrity, those loads that do significantly affect burst or collapse shall be determined and assessed in combination with the loads due to pressure with a safety factor of 1.2 on the combined primary loads and 1.0 on axial secondary loads.

5.5 Programs and Manuals

2. Accident induced leakage performance criterion: The primary to secondary accident induced leakage rate for any design basis accident, other than a SG tube rupture, shall not exceed the leakage rate assumed in the accident analysis in terms of total leakage rate for all SGs and leakage rate for an individual SG. Leakage is not to exceed 100 gpd per SG.

3. The operational LEAKAGE performance criterion is specified in LCO 3.4.13, "RCS Operational LEAKAGE."

c. Provisions for SG tube plugging criteria. Tubes found by inservice inspection to contain flaws with a depth equal to or exceeding 40% of the nominal tube wall thickness shall be plugged.

d. Provisions for SG tube inspections. Periodic SG tube inspections shall be performed. The number and portions of the tubes inspected and methods of inspection shall be performed with the objective of detecting flaws of any type (e.g., volumetric flaws, axial, and circumferential cracks) that may be present along the length of the tube, from the tube-to-tubesheet weld at the tube inlet to the tube-to-tubesheet weld at the tube outlet, and that may satisfy the applicable tube plugging criteria. The tube-to-tubesheet weld is not part of the tube. In addition to meeting the requirements of d.1, d.2, and d.3 below, the inspection scope, inspection methods, and inspection intervals shall be such as to ensure that SG tube integrity is maintained until the next SG inspection. A degradation assessment shall be performed to determine the type and location of flaws to which the tubes may be susceptible and, based on this assessment, to determine which inspection methods need to be employed and at what locations.

1. Inspect 100% of the tubes in each SG during the first refueling outage following SG installation.

2. After the first refueling outage following SG installation, inspect each SG at least every 72

5.5 Programs and Manuals

effective full power months or at least every third refueling outage (whichever results in more frequent inspections). In addition, the minimum number of tubes inspected at each scheduled inspection shall be the number of tubes in all SGs divided by the number of SG inspection outages scheduled in each inspection period as defined in a, b, c, and d below. If a degradation assessment indicates the potential for a type of degradation to occur at a location not previously inspected with a technique capable of detecting this type of degradation at this location and that may satisfy the applicable tube plugging criteria, the minimum number of locations inspected with such a capable inspection technique during the remainder of the inspection period may be prorated. The fraction of locations to be inspected for this potential type of degradation at this location at the end of the inspection period shall be no less than the ratio of the number of times the SG is scheduled to be inspected in the inspection period after the determination that a new form of degradation could potentially be occurring at this location divided by the total number of times the SG is scheduled to be inspected in the inspection period. Each inspection period defined below may be extended up to 3 effective full power months to include a SG inspection outage in an inspection period and the subsequent inspection period begins at the conclusion of the included SG inspection outage.

a) After the first refueling outage following SG installation, inspect 100% of the tubes during the next 144 effective full power months. This constitutes the first inspection period;

b) During the next 120 effective full power months, inspect 100% of the tubes. This constitutes the second inspection period;

5.5 Programs and Manuals

c) During the next 96 effective full power months, inspect 100% of the tubes. This constitutes the third inspection period; and

d) During the remaining life of the SGs, inspect 100% of the tubes every 72 effective full power months. This constitutes the fourth and subsequent inspection periods.

3. If crack indications are found in any SG tube, then the next inspection for each affected and potentially affected SG for the degradation mechanism that caused the crack indication shall not exceed 24 effective full power months or one refueling outage (whichever results in more frequent inspection). If definitive information, such as from examination of a pulled tube, diagnostic non-destructive testing, or engineering evaluation indicates that a crack-like indication is not associated with a crack(s), then the indication need not be treated as a crack.

e. Provisions for monitoring operational primary to secondary LEAKAGE.

5.5.10 Secondary Water Chemistry Program

This program provides controls for monitoring secondary water chemistry to inhibit steam generator tube degradation and low pressure turbine disc stress corrosion cracking. The program shall include:

a. Identification of a sampling schedule for the critical variables and control points for these variables;

b. Identification of the procedures used to measure the values of the critical variables;

5.5 Programs and Manuals

c. Identification of process sampling points which shall include monitoring the discharge of the condensate pumps for evidence of condenser in leakage;

d. Procedures for the recording and management of data;

e. Procedures defining corrective actions for all off control point chemistry conditions; and

f. A procedure identifying the authority responsible for the interpretation of the data and the sequence and timing of administrative events, which are required to initiate corrective action.

5.5.11 Ventilation Filter Testing Program

A program shall be established to implement the following required testing of engineered safety feature (ESF) filter ventilation systems. Tests described in Specifications 5.5.11.a and 5.5.11.b shall be performed once per 18 months for ventilation systems other than the Iodine Removal System (IRS) and 24 months for the IRS; after each complete or partial replacement of the high efficiency particulate air (HEPA) filter bank or charcoal adsorber bank; after any structural maintenance on the HEPA filter or charcoal adsorber housing; and following painting, fire, or chemical release in any ventilation zone communicating with the system.

Tests described in Specification 5.5.11.c shall be performed once per 18 months for ventilation systems other than the IRS and 24 months for the IRS; after 720 hours of system operation; after any structural maintenance on the HEPA filter or charcoal adsorber housing; and following painting, fire, or chemical release in any ventilation zone communicating with the system.

Tests described in Specification 5.5.11.d shall be performed once per 18 months for ventilation systems other than the IRS and 24 months for the IRS.

5.5 Programs and Manuals

The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the Ventilation Filter Testing Program test frequencies.

a. Demonstrate for each of the ESF systems that an inplace test of the HEPA filters shows a penetration and system bypass ≤ 1.0% (≤0.05% for the CREVS only) when tested in accordance with Regulatory Positions C.5.a and C.5.c of Regulatory Guide 1.52, Revision 2, and ANSI N510-1975, at the system flowrate specified as follows ± 10%:

ESF Ventilation System	Flowrate
Control Room Emergency Ventilation System (CREVS)	10,000 cfm
Penetration Room Exhaust Ventilation System (PREVS)	2,000 cfm
IRS	20,000 cfm

b. Demonstrate for each of the ESF systems that an inplace test of the charcoal adsorber shows a penetration and system bypass ≤ 1.0% when tested in accordance with Regulatory Positions C.5.a and C.5.d of Regulatory Guide 1.52, Revision 2, and ANSI N510-1975, at the system flowrate specified as follows ± 10%:

ESF Ventilation System	Flowrate
CREVS	10,000 cfm
PREVS	2,000 cfm
IRS	20,000 cfm

c. Demonstrate for each of the ESF systems within 31 days after removal that a laboratory test of a sample of the charcoal adsorber, when obtained as described in Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, shows the methyl iodide penetration less than the value specified below when tested in accordance with ASTM D3803-1989 at a

5.5 Programs and Manuals

temperature of 30°C and greater than or equal to the relative humidity specified as follows:

ESF Ventilation System	Penetrations	RH
CREVS	4.5%	70%
PREVS	34.5%	95%
IRS	34.5%	95%

d. For each of the ESF systems, demonstrate the pressure drop across the combined HEPA filters, the prefilters, and the charcoal adsorbers is less than the value specified below when tested in accordance with Regulatory Guide 1.52, Revision 2, and ANSI N510-1975 at the system flowrate specified as follows ± 10%:

ESF Ventilation System	Delta P	Flowrate
CREVS	6 inwg	10,000 cfm
PREVS	6 inwg	2,000 cfm
IRS	6 inwg	20,000 cfm

5.5.12 Explosive Gas and Storage Tank Radioactivity Monitoring Program

This program provides control for potentially explosive gas mixtures contained in the Waste Gas Holdup System and the quantity of radioactivity contained in gas storage tanks. The gaseous radioactivity quantities shall be determined following the methodology in the ODCM.

The program shall include:

a. The limits for concentrations of oxygen in the Waste Gas Holdup System and a surveillance program to ensure the limits are maintained. Such limits shall be appropriate to the system's design criteria (i.e., whether or not the system is designed to withstand a hydrogen explosion); and

5.5 Programs and Manuals

b. A surveillance program to ensure that the quantity of radioactivity contained in each gas storage tank is less than or equal to 58,500 curies noble gases (considered as Xe-133).

The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the Explosive Gas and Storage Tank Radioactivity Monitoring Program surveillance Frequencies.

5.5.13 Diesel Fuel Oil Testing Program

A Diesel Fuel Oil Testing Program to implement required testing of both new fuel oil and stored fuel oil shall be established. The program shall include sampling and testing requirements, and acceptance criteria, all in accordance with ASTM Standards. The purpose of the program is to establish the following:

a. Acceptability of new fuel oil for use prior to addition to storage tanks by determining that the fuel oil has:

   1. An American Petroleum Institute gravity or an absolute specific gravity within limits,

   2. A flash point and kinematic viscosity within limits for ASTM 2D fuel oil, and

   3. Water and sediment $\leq$ 0.05%.

b. Within 31 days following addition of new fuel oil to the storage tanks, verify that the properties of the new fuel oil, other than those addressed in a., above, are within limits for ASTM 2D fuel oil; and

c. Total particulate concentration of the fuel oil, when determined by gravimetric analysis based on ASTM D2276-1989, is $\leq$ 10 mg/l when tested every 92 days.

d. The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the Diesel Fuel Oil Testing Frequencies.

5.5 Programs and Manuals

5.5.14 Technical Specifications Bases Control Program

This program provides a means for processing changes to the Bases of these Technical Specifications.

a. Changes to the Bases of the Technical Specifications shall be made under appropriate administrative controls and reviews.

b. Licensees may make changes to Bases without prior NRC approval provided the changes do not require either of the following:

   1. A change in the Technical Specifications incorporated in the license; or

   2. A change to the UFSAR or Bases that requires NRC approval pursuant to 10 CFR 50.59.

c. The Bases Control Program shall contain provisions to ensure that the Bases are maintained consistent with the UFSAR.

d. Proposed changes that meet the criteria of Specification 5.5.14b above shall be reviewed and approved by the NRC prior to implementation. Changes to the Bases implemented without prior NRC approval shall be provided to the NRC on a frequency consistent with 10 CFR 50.71(e).

5.5.15 Safety Function Determination Program (SFDP)

This program ensures loss of safety function is detected and appropriate actions taken. Upon entry into Limiting Condition for Operation (LCO) 3.0.6, an evaluation shall be made to determine if loss of safety function exists. Additionally, other appropriate limitations and remedial or compensatory actions may be identified to be taken as a result of the support system inoperability and corresponding exception to entering supported system Condition and Required Actions. This program implements the requirements of LCO 3.0.6. The SFDP shall contain the following:

5.5 Programs and Manuals

a. Provisions for cross-train checks to ensure a loss of the capability to perform the safety function assumed in the accident analysis does not go undetected;

b. Provisions for ensuring the plant is maintained in a safe condition if a loss of function condition exists;

c. Provisions to ensure that an inoperable supported system's Completion Time is not inappropriately extended as a result of multiple support system inoperabilities; and

d. Other appropriate limitations and remedial or compensatory actions.

A loss of safety function exists when, assuming no concurrent single failure, a safety function assumed in the accident analysis cannot be performed. For the purpose of this program, a loss of safety function may exist when a support system is inoperable, and:

a. A required system redundant to system(s) supported by the inoperable support system is also inoperable; or

b. A required system redundant to system(s) in turn supported by the inoperable supported system is also inoperable; or

c. A required system redundant to support system(s) for the supported systems (a) and (b) above is also inoperable.

The SFDP identifies where a loss of safety function exists. If a loss of safety function is determined to exist by this program, the appropriate Conditions and Required Actions of the LCO in which the loss of safety function exists are required to be entered.

5.5 Programs and Manuals

5.5.16 Containment Leakage Rate Testing Program

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

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

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

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

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

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

Leakage rate acceptance criteria are:

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

5.5 Programs and Manuals

b. Air lock testing acceptance criteria are:

1. Overall air lock leakage rate is $\leq 0.05\ L_a$ when tested at $\geq P_a$.

2. For each door, leakage rate is $\leq 0.0002\ L_a$ when pressurized to ≥ 15 psig.

The provisions of SR 3.0.2 do not apply to the test frequencies specified in the Containment Leakage Rate Testing Program.

The provisions of SR 3.0.3 are applicable to the Containment Leakage Rate Testing Program.

5.5.17 <u>Control Room Envelope Habitability Program</u>

A Control Room Envelope (CRE) Habitability Program shall be established and implemented to ensure that CRE habitability is maintained such that, with an OPERABLE Control Room Emergency Ventilation System (CREVS), CRE occupants can control the reactor safely under normal conditions and maintain it in a safe condition following a radiological event, hazardous chemical release, or a smoke challenge. The program shall ensure that adequate radiation protection is provided to permit access and occupancy of the CRE under design basis accident (DBA) conditions without personnel receiving radiation exposures in excess of 5 rem whole body or its equivalent to any part of the body for the duration of the accident. The program shall include the following elements:

a. The definition of CRE and the CRE boundary.

b. Requirements for maintaining CRE boundary in its design condition including configuration control and preventive maintenance.

c. Requirements for (i) determining the unfiltered air inleakage past the CRE boundary into the CRE in accordance with the testing methods and at the Frequencies specified in Sections C.1 and C.2 of Regulatory Guide 1.197, "Demonstrating Control

5.5 Programs and Manuals

Room Envelope Integrity at Nuclear Power Reactors," Revision 0, May 2003, and (ii) assessing CRE habitability at the Frequencies specified in Sections C.1 and C.2 of Regulatory Guide 1.197, Revision 0.

d. License controlled programs will be used to verify the integrity of the CRE boundary. Conditions that generate relevant information from those programs will be entered into the corrective action process and shall be trended and used as part of the 36 month assessments of the CRE boundary.

e. The quantitative limits on unfiltered air inleakage into the CRE. These limits shall be stated in a manner to allow direct comparison to the unfiltered air inleakage measured by the testing described in paragraph c. The unfiltered air inleakage limit for radiological challenges is the inleakage flow rate assumed in the licensing basis analyses of DBA consequences. Unfiltered air inleakage limits for hazardous chemicals must ensure that exposure of CRE occupants to these hazards will be within the assumptions in the licensing basis.

f. The provisions of SR 3.0.2 are applicable to the Frequencies for assessing CRE habitability, determining CRE unfiltered inleakage, and assessing the CRE boundary as required by paragraphs c and d respectively.

5.6 Reporting Requirements

a. The scope of inspections performed on each SG,

b. Degradation mechanisms found,

c. Nondestructive examination techniques utilized for each degradation mechanism,

d. Location, orientation (if linear), and measured sizes (if available) of service induced indications,

e. Number of tubes plugged during the inspection outage for each degradation mechanism,

f. The number and percentage of tubes plugged to date and the effective plugging percentage in each steam generator,

g. The results of condition monitoring, including the results of tube pulls and in-situ testing.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO

AMENDMENT NO. 308 TO RENEWED FACILITY OPERATING LICENSE NO. DPR-53

AMENDMENT NO. 286 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 October 31, 2013 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML13308A387), Calvert Cliffs Nuclear Power Plant, LLC, the licensee, submitted a request to the Nuclear Regulatory Commission (NRC) for changes to the Calvert Cliffs Nuclear Power Plant Unit Nos. 1 and 2 (Calvert Cliffs) Technical Specifications (TSs). The NRC staff published a proposed no significant hazards consideration determination in the *Federal Register* (FR) on July 22, 2014 (79 FR 42547).

The proposed changes revise TSs associated with implementation issues related to steam generator (SG) inspection periods. The proposed changes are consistent with Commission-approved Technical Specifications Task Force (TSTF) Standard Technical Specifications (STS) Change Traveler 510, Revision 2, "Revision to Steam Generator Program Inspection Frequencies and Tube Sample Selection" (ADAMS Accession No. ML110610350).

2.0 BACKGROUND

The TSTF Travelers, such as TSTF-510, evaluate changes to the STS. The STS applicable to the Calvert Cliffs Nuclear Steam Supply System is NUREG-1432, "Standard Technical Specifications - Combustion Engineering Plants." The current STS provisions related to SG programs were established in May 2005 with the NRC staff's approval of TSTF-449, Revision 4, "Steam Generator Tube Integrity" (NRC *Federal Register* Notice of Availability of May 6, 2005, (70 FR 24126)). The TSTF-449 changes to the STS incorporated a new, largely performance-based approach for ensuring that the integrity of the SG tubes is maintained. The performance-based provisions were supplemented by prescriptive provisions relating to tube inspections and tube repair limits to ensure that conditions adverse to quality are detected and corrected on a timely basis. By letter dated March 9, 2006 (ADAMS Accession No. ML060460089), the NRC approved TSTF-449 for implementation in the Calvert Cliffs TS.

Enclosure

After the issuance of TSTF-449, TSTF-510 was developed to reflect the industry's early implementation experience with respect to TSTF-449. The TSTF-510 characterizes the changes as editorial corrections, changes, and clarifications intended to improve internal consistency, consistency with implementing industry documents, and usability without changing the intent of the requirements. Further, the proposed changes are an improvement to the existing SG inspection requirements and continue to provide assurance that the plant licensing basis will be maintained between SG inspections.

3.0 REGULATORY REQUIREMENTS

3.1 Description of System

The SG tubes in pressurized-water reactors (PWRs) have a number of important safety functions. These tubes are an integral part of the reactor coolant pressure boundary (RCPB) and, as such, are relied upon to maintain primary system pressure and inventory. As part of the RCPB, the SG tubes are unique in that they are also relied upon as a heat transfer surface between the primary and secondary systems such that residual heat can be removed from the primary system and are relied upon to isolate the radioactive fission products in the primary coolant from the secondary system. In addition, the SG tubes are relied upon to maintain their integrity to be consistent with the containment objectives of preventing uncontrolled fission product release under conditions resulting from core damage during severe accidents.

3.2 Regulatory Requirements and Guidance

The construction permits for Calvert Cliffs 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 effective on May 21, 1971. As stated in SECY-92-223, dated September 18, 1992, (ADAMS Accession No. ML003763736), the Commission decided not to apply the Appendix A GDC to plants with construction permits issued prior to May 21, 1971. The Calvert Cliffs Updated Final Safety Analysis Report (UFSAR) states that the plant was designed and constructed to meet the intent of the GDC published in July 1967. The plant GDC is discussed in the UFSAR Appendix 1C, "AEC Proposed General Design Criteria for Nuclear Power Plants."

The GDC in Appendix A to 10 CFR Part 50 state that the RCPB shall have "an extremely low probability of abnormal leakage...and of gross rupture" (GDC 14); "shall be designed with sufficient margin to assure that the design conditions...are not exceeded..." (GDC 15); "shall be designed with sufficient margin that when stressed ...(1) the boundary behaves in a non-brittle manner and (2) the probability of rapidly propagating fracture is minimized" (GDC 31); shall be of "the highest quality standards practical" (GDC 30); and shall be designed to permit "periodic inspection and testing...to assess...structural and leaktight integrity" (GDC 32).

Paragraph 50.55a(c)(1) of 10 CFR specifies that components which are part of the RCPB must meet the requirements for Class 1 components in Section III of the American Society of Mechanical Engineers *Boiler and Pressure Vessel Code* (ASME Code). Paragraph 50.55a(g)(4) of 10 CFR further requires, in part, that throughout the service life of a PWR facility, ASME Code Class 1 components meet the requirements, except design and access provisions and pre-service examination requirements in Section XI, "Rules for Inservice Inspection [ISI] of Nuclear Power Plant Components," of the ASME Code, to the extent practical. This requirement includes the inspection and repair criteria of Section XI of the ASME Code.

The regulations at 10 CFR 50.36, "Technical specifications," establishes the requirements related to the content of the TS. Pursuant to 10 CFR 50.36, TSs are required to include items in the following five categories related to station operation: (1) safety limits, limiting safety system settings, and limiting control settings; (2) limiting conditions of operation (LCOs); (3) surveillance requirements (SRs); (4) design features; and (5) administrative controls. As described in TSTF-510, LCOs and accompanying action statements and SRs in the STS relevant to SG tube integrity are in TS 3.4.13, "Reactor Coolant System Operational Leakage," and TS 3.4.18 (SR 3.4.18.2), "Steam Generator (SG) Tube Integrity." The SRs, in the SG tube integrity TS, reference the SG program which is defined in the STS administrative controls. In its application, the licensee stated "Calvert Cliffs uses different numbering than the Improved Standard Technical Specifications in one instance. The Steam Generator Tube Inspection Report is TS 5.6.9 in the Calvert Cliffs TS." The Steam Generator Tube Inspection Report is numbered as TS 5.6.7 in the Improved Standard Technical Specifications.

Paragraph 50.36(c)(5) of 10 CFR defines administrative controls as "the provisions relating to organization and management, procedures, recordkeeping, review and audit, and reporting necessary to assure the operation of the facility in a safe manner." Programs established by the licensee to operate the facility in a safe manner, including the SG Program, are listed in the administrative controls section of the TS. For Calvert Cliffs, the SG Program is defined in TS 5.5.9, while the reporting requirements relating to implementation of the SG Program are in TS 5.6.9.

The TS 5.5.9 requires that an SG program be established and implemented to ensure that SG tube integrity is maintained. The TS 5.5.9.a requires that a condition monitoring assessment be performed during each outage in which the SG tubes are inspected, to confirm that the performance criteria are being met. The SG tube integrity is maintained by meeting the performance criteria specified in TS 5.5.9.b for structural and leakage integrity, consistent with the plant design and licensing basis. The applicable tube repair criteria, specified in TS 5.5.9.c, are that tubes found during ISI to contain flaws with a depth equal to or exceeding 40 percent of the nominal wall thickness shall be plugged. The TS 5.5.9.d includes provisions regarding the scope, frequency, and methods of SG tube inspections. These provisions require that the inspections be performed with the objective of detecting flaws of any type that: (1) may be present along the length of a tube, from the tube-to-tubesheet weld at the tube inlet to the tube-to-tubesheet weld at the tube outlet; and (2) may satisfy the applicable tube repair criteria.

4.0 TECHNICAL EVALUATION

Each proposed change to the TS is described individually below, followed by the NRC staff's assessment of the change.

4.1 TS 5.5.9, "Steam Generator (SG) Program"

The last sentence in the introductory paragraph of Calvert Cliffs TS 5.5.9 currently states, "In addition, the Steam Generator Program shall include the following provisions."

The sentence is revised to say "In addition, the Steam Generator Program shall include the following: " The reason for this change is that the subsequent paragraph starts with "Provisions for" and stating "provisions" in the introductory paragraph is duplicative.

The NRC staff reviewed the licensee's proposed change to TS 5.5.9 and has determined that the word "provisions" in the introductory paragraph is duplicative. The NRC staff finds that the change is administrative in nature, and therefore, is acceptable.

4.2 <u>TS 5.5.9, Paragraph 5.5.9.b.1, "Structural integrity performance criterion"</u>

The first sentence of Calvert Cliffs TS 5.5.9.b.1 currently states:

> All in-service steam generator tubes shall retain structural integrity over the full range of normal operating conditions (including startup, operation in the power range, hot standby, and cool down, and all anticipated transients included in the design specification) and design basis accidents.

The licensee proposed to revise the sentence as follows:

> All in-service steam generator tubes shall retain structural integrity over the full range of normal operating conditions (including startup, operation in the power range, hot standby, and cool down), all anticipated transients included in the design specification, and design-basis accidents.

The basis for the change is that the sentence inappropriately includes anticipated transients in the description of normal operating conditions.

The NRC staff has determined that the current wording is incorrect and that anticipated transients should be differentiated from normal operating conditions. Therefore, the NRC staff finds that the change is acceptable.

4.3. <u>Paragraph 5.5.9.c, "Provisions for SG tube repair criteria," Paragraph 5.5.9.d, "Provisions for SG tube inspections," TS 3.4.18, "Steam Generator (SG) Tube Integrity," and SR 3.4.18.2 for LCO 3.4.18</u>

The licensee proposed to change all references to the words "tube repair criteria" to "tube plugging criteria." According to the licensee, this change is intended to be consistent with the treatment of SG tube repair throughout TS 5.5.9.c.

The NRC staff determined that the proposed revision adds clarity to the specification because generally, one of two actions must be taken when the criteria are exceeded. One action is to remove the tube from service by plugging the tube at both tube ends. The alternative action is to repair the tube, but only if such a repair is permitted in the TS by paragraph 5.5.9.d. Calvert Cliffs does not have any approved alternate repair criteria, and thus, plugging is the only available option if the criteria are exceeded. Therefore, the NRC staff finds that the changes are acceptable.

4.4 Paragraph 5.5.9.d, "Provisions for SG tube inspection"

The licensee proposed to change the term "assessment of degradation" to "degradation assessment" to be consistent with the terminology used in industry program documents.

The NRC staff agrees that the terminology should be consistent. Further, the proposed wording does not involve a technical change to the specification. Therefore, the NRC staff finds that the change is acceptable.

4.5 Paragraph 5.5.9.d.1

The paragraph currently states, "Inspect 100% of the tubes in each SG during the first refueling outage following SG replacement."

The proposed change would replace "SG replacement" with "SG installation." The basis for the change is that it will allow the SG program to apply to both existing plants and new plants.

The NRC staff agrees that the SG program can apply to both existing and new plants. The change in wording allows for consistency between Calvert Cliffs and other plants. Since this wording modification does not involve any technical or functional change for Calvert Cliffs, the NRC staff finds it acceptable.

4.6 Paragraph 5.5.9.d.2 (SGs with alloy 690 thermally treated tubes)

The TSTF-510 is written to accommodate plants with several variations of SG tubing material. As described in the Calvert Cliffs UFSAR, Revision 45, Table 4-10, "Material exposed to coolant," the Calvert Cliffs SGs employ a thermally treated (TT) alloy 690 tubing design.

Paragraph 5.5.9.d.2 of the Calvert Cliffs TS currently states:

> Inspect 100% of the tubes at sequential periods of 144, 108, 72, and, thereafter, 60 effective full power months. The first sequential period shall be considered to begin after the first inservice inspection of the SGs. In addition, inspect 50% of the tubes by the refueling outage nearest the midpoint of the period and the remaining 50% by the refueling outage nearest the end of the period. No SG shall operate for more than 72 effective full power months or three refueling outages (whichever is less) without being inspected.

The licensee proposed to replace paragraph 5.5.9.d.2 with the following insert:

> After the first refueling outage following SG installation, inspect each SG at least every 72 effective full power months or at least every third refueling outage (whichever results in more frequent inspections). In addition, the minimum number of tubes inspected at each scheduled inspection shall be the number of tubes in all SGs divided by the number of SG inspection outages scheduled in each inspection period as defined in a, b, c, and d below. If a degradation assessment indicates the potential for a type of degradation to occur at a location not previously inspected with a technique capable of detecting this type of degradation at this location and that may satisfy the applicable tube plugging

criteria, the minimum number of locations inspected with such a capable inspection technique during the remainder of the inspection period may be prorated. The fraction of locations to be inspected for this potential type of degradation at this location at the end of the inspection period shall be no less than the ratio of the number of times the SG is scheduled to be inspected in the inspection period after the determination that a new form of degradation could potentially be occurring at this location divided by the total number of times the SG is scheduled to be inspected in the inspection period. Each inspection period defined below may be extended up to 3 effective full power months to include a SG inspection outage in an inspection period and the subsequent inspection period begins at the conclusion of the included SG inspection outage.

a) After the first refueling outage following SG installation, inspect 100% of the tubes during the next 144 effective full power months. This constitutes the first inspection period;

b) During the next 120 effective full power months, inspect 100% of the tubes. This constitutes the second inspection period;

c) During the next 96 effective full power months, inspection [sic] 100% of the tubes. This constitutes the third inspection period; and

d) During the remaining life of the SGs, inspect 100% of the tubes every 72 effective full power months. This constitutes the fourth and subsequent inspection periods.

The licensee proposed, in part, to move the first two sentences of paragraph 5.5.9.d.2 to the inspection periods, as specified in subsections a, b, c, and d of the revised paragraph, and make editorial changes to improve clarity. The NRC staff has determined that these changes are clarifying in nature and do not change the current intent of these two sentences. However, the licensee's application also includes three proposed changes to when inspections are performed as follows:

- The second inspection period would be revised from 108 to 120 effective full power months (EFPM).
- The third inspection period would be revised from 72 to 96 EFPM.
- The fourth and subsequent inspection periods would be revised from 60 to 72 EFPM.

The licensee characterized these changes as marginal increases for consistency with typical fuel cycle lengths that better accommodate the scheduling of inspections. The NRC staff observes that depending on the actual plant inspection schedule, these changes could impact the number of inspections in a given period, as well as the sample size. However, inspection sample sizes will continue to be subject to paragraph 5.5.9.d.2, which states that in addition to meeting the requirements of paragraph 5.5.9.d.2, the inspection scope, inspection methods, and inspection intervals shall be such as to ensure SG tube integrity is maintained until the next scheduled inspection. Therefore, the NRC staff has determined that, with the proposed changes to the length of the second and subsequent inspection periods, compliance with the

SG program requirements in TS 5.5.9.d.2 will continue to ensure both adequate inspection scopes and tube integrity for the reasons addressed below.

For each inspection period, paragraph 5.5.9.d.2 currently requires that at least 50 percent of the tubes be inspected by the refueling outage nearest to the mid-point of the inspection period and the remaining 50 percent by the refueling outage nearest the end of the inspection period. The NRC staff notes that if there are not an equal number of inspections in the first half and second half of the inspection period, the average minimum sampling requirement may be markedly different for inspections in the first half of the inspection period, as compared to those in the second half, even when there are uniform intervals between each inspection.

For example, a hypothetical plant in the second (120 EFPM) inspection period with a scheduled 36-month interval (two 18-month fuel cycles) between each inspection would currently be required to inspect 50 percent of the tubes by the refueling outage nearest the midpoint of the inspection period, which would be the third refueling outage in the period (after 54 EFPM), 6 months before the mid-point (assuming an inspection was performed at the very end of the 144 EFPM inspection period). However, since no inspection is scheduled for that outage (because inspections take place every other outage – once every 36 months), then the full 50 percent sample must be performed during the inspection scheduled for the second refueling outage in the period. Two inspections would be scheduled to occur in the second half of the inspection period, at 72 and 108 months into the inspection period. Thus, the current sampling requirement could be satisfied by performing a 25 percent sample during each of these inspections or other combinations of sampling (e.g., 10 percent during one and 40 percent in the other) totaling 50 percent.

The NRC staff has determined that there is no basis for the minimum initial sample size to potentially have to vary so much from inspection to inspection. The licensee proposed to revise this requirement such that the minimum sample size for a given inspection in a given inspection period is 100 percent divided by the number of scheduled inspections during that inspection period. For the above example, the proposed change would result in an uniform initial minimum sample size of 33.3 percent for each of the three scheduled inspections during the inspection period. The NRC staff has determined that this proposed revision is an improvement to the existing requirement, since it provides a more consistent minimum initial sampling requirement.

The proposed third and fourth sentences of paragraph 5.5.9.d.2 state, "If a degradation assessment indicates the potential for a type of degradation to occur at a location not previously inspected with a technique capable of detecting this type of degradation at this location and that may satisfy the applicable tube repair criteria, the minimum number of locations inspected with such a capable inspection technique during the remainder of the inspection period may be prorated. The fraction of locations to be inspected for this potential type of degradation at this location at the end of the inspection period shall be no less than the ratio of the number of times the SG is scheduled to be inspected in the inspection period after the determination that a new form of degradation could potentially be occurring at this location divided by the total number of times the SG is scheduled to be inspected in the inspection period." This addresses the possibility that a degradation assessment in accordance with paragraph 5.5.9.d.2 will indicate that the tubing may be susceptible to a type of degradation at a location not previously inspected with a technique capable of detecting that type of degradation at that location. For example, new information from another similar plant becomes available, indicating the potential for circumferential cracking at a specific location on the tube. Previous degradation assessments had not identified the potential for this type of degradation at this location. Thus, previous inspections of this location had not been performed with a technique capable of detecting circumferential cracks. However, now that the potential for circumferential cracking has been identified at this location, paragraph 5.5.9.d.2 requires an inspection with a method capable of detection of a crack that may satisfy the applicable tube repair criteria.

Furthermore, if this inspection is performed for the first time during the third of four SG inspections scheduled for the 144 EFPM inspection period, the current paragraph 5.5.9.d.2 does not specifically identify whether 100 percent of the tubes at this location need to be inspected by the end of the 144 EFPM inspection period, or whether a prorated approach may be taken. The NRC staff addressed this question in Issue 1 of NRC Regulatory Information Summary (RIS) 2009-04, "Steam Generator Tube Inspection Requirements," dated April 3, 2009 (ADAMS Accession No. ML083470557), as follows:

> **Issue 1**: *A licensee may identify a new potential degradation mechanism after the first inspection in a sequential period. If this occurs, what are the expectations concerning the scope of examinations for this new potential degradation mechanism for the remainder of the period (e.g., do 100 percent of the tubes have to be inspected by the end of the period or can the sample be prorated for the remaining part of the period)?*
>
> [NRC Staff Position:] The TS contain requirements that are a mixture of prescriptive and performance-based elements. Paragraph "d" of these requirements indicates that the inspection scope, inspection methods, and inspection intervals shall be sufficient to ensure that SG tube integrity is maintained until the next SG inspection. Paragraph "d" is a performance-based element because it describes the goal of the inspections but does not specify how to achieve the goal. However, paragraph "d.2" is a prescriptive element because it specifies that the licensee must inspect 100 percent of the tubes at specified periods.
>
> If an assessment of degradation performed after the first inspection in a sequential period results in a licensee concluding that a new degradation mechanism (not anticipated during the prior inspections in that period) may potentially occur, the scope of inspections in the remaining portion of the period should be sufficient to ensure SG tube integrity for the period between inspections.
>
> In addition, to satisfy the prescriptive requirements of paragraph "d.2" that the licensee must inspect 100 percent of the tubes within a specified period, a prorated sample for the remaining portion of the period is appropriate for this potentially new degradation mechanism. This prorated sample should be such that if the licensee had implemented it at the beginning of the period, the TS requirement for the 100 percent inspection in the entire period (for this degradation mechanism) would have been met. A prorated sample is appropriate because (1) the licensee would have performed the prior inspections in this sequential period consistently with the requirements, and (2) the scope of inspections must be sufficient to ensure that the licensee maintains SG tube integrity for the period between inspections.

The NRC staff finds that relocation of information in sentences 3 and 4, as described above, clarifies the existing requirement, such that it is consistent with the NRC staff's position from RIS 2009-04, and therefore, is acceptable.

The proposed fifth sentence in paragraph 5.5.9.d.2 states, "Each inspection period defined below may be extended up to 3 effective full power months to include a SG inspection outage in an inspection period and the subsequent inspection period begins at the conclusion of the included SG inspection outage." Allowing extension of the inspection periods by up to an additional 3 EFPM potentially impacts the average tube inspection sample size to be implemented during a given inspection in that period. For example, if four SG inspections are scheduled to occur within the nominal 144 EFPM period, the minimum sample size for each of the four inspections could average as little as 25 percent of the tube population. If a fifth inspection can be included within the period by extending the period by 3 EFPM, then the minimum sample size for each of the five inspections could average as little as 20 percent of the tube population. Since the subsequent period begins at the end of the included SG inspection outage, the proposed change does not impact the required frequency of SG inspection.

Required tube inspection sample sizes are also subject to the performance-based requirement in paragraph 5.5.9.d.2 which states, in part, that in addition to meeting the requirements of paragraph 5.5.9.d.2, "the inspection scope, inspection methods, and inspection intervals shall be such as to ensure that SG tube integrity is maintained until the next scheduled SG inspection." This requirement remains unchanged under the proposal. The NRC staff has determined that the proposed fifth sentence, by allowing the potential for smaller sample sizes, involves only a relatively minor relaxation to the existing sampling requirements in paragraph 5.5.9.d.2. However, the performance-based requirements in 5.5.9.d.2 ensure that adequate inspection sampling will be performed to ensure tube integrity is maintained. Thus, the NRC staff finds that the proposed change is acceptable.

Finally, the first sentence of the proposed revision to paragraph 5.5.9.d.2, "After the first refueling outage following SG installation, inspect each SG at least every 72 effective full power months or at least every third refueling outage (whichever results in more frequent inspections)," replaces the last sentence of the current paragraph 5.5.9.d.2, "No SG shall operate for more than 72 effective full power months or three refueling outages (whichever is less) without being inspected." Both versions establish the minimum allowable SG inspection frequency as at least every 72 EFPM or at least every third refueling outage (whichever results in more frequent inspections). This minimum inspection frequency in the proposed version is unchanged from the current requirement in the Calvert Cliffs TSs. The NRC staff has determined that the wording changes in the sentence are of an editorial and clarifying nature and are not material, such that the current intent of the requirement is unchanged. Thus, the NRC staff finds that the proposed change is acceptable.

4.7 Paragraph 5.5.9.d.3

The first sentence of paragraph 5.5.9.d.3 of the Calvert Cliffs TS currently states:

> If crack indications are found in any SG tube, then the next inspection for each SG for the degradation mechanism that caused the crack indication shall not exceed 24 effective full power months or one refueling outage (whichever is less).

The licensee proposed to revise this sentence as follows:

> If crack indications are found in any SG tube, then the next inspection for each affected and potentially affected SG for the degradation mechanism that caused the crack indication shall not exceed 24 effective full power months or one refueling outage (whichever results in more frequent inspections).

The proposed change is replacing the words "for each SG" with words "for each affected and potentially affected SG." The licensee stated that the change should be made because the existing wording can be misinterpreted.

The proposed changes in paragraph 5.5.9.d.3 permits SG inspection intervals to extend over multiple fuel cycles for SGs with alloy 600 TT and 690 TT tubing, assuming that such intervals can be implemented while ensuring tube integrity is maintained in accordance with paragraph 5.5.9.d. However, stress-corrosion cracks may not become detectable by inspection until the crack depth approaches the tube repair limit. In addition, stress-corrosion cracks may exhibit high growth rates. For these reasons, once cracks have been found in any SG tube, paragraph 5.5.9.d.3 restricts the allowable interval to the next scheduled inspection to 24 EFPM or one refueling outage (whichever is less). The intent of this requirement is that it applies to the affected SG and to any other SG, which may be potentially affected by the degradation mechanism that caused the known crack(s).

For example, a root cause analysis in response to the initial finding of one or more cracks might reveal that the crack(s) are associated with a manufacturing anomaly which causes locally high residual stress, which in turn, caused the early initiation of cracks at the affected locations. If it can be established that the extent of condition of the manufacturing anomaly applies only to one SG and not the others, then the NRC staff agrees that only the affected SG needs to be inspected within 24 EFPM or one refueling cycle in accordance with paragraph 5.5.9.d.3. Conversely, if it cannot be established that the manufacturing anomaly applies to just one SG, then all potentially affected SGs would have to be inspected. The next scheduled inspections of the other SGs will continue to be subject to all other provisions of paragraph 5.5.9.d.3. The NRC staff finds that the proposed change to paragraph 5.5.9.d.3 is acceptable, because it clarifies the intent the paragraph.

4.8 <u>Specification 5.6.9, "Steam Generator Tube Inspection Report"</u>

This specification lists items a. through h. to be included in a report which shall be submitted within 180 days after the average reactor coolant temperature exceeds 200 °F [degrees Fahrenheit] following completion of an inspection performed in accordance with TS 5.5.9 "Steam Generator (SG) Program."

Item b of the Calvert Cliffs TS 5.6.9 currently reads: "Active degradation mechanisms found." The proposed revision reads: "Degradation mechanisms found."

Item e of the Calvert Cliffs TS 5.6.9 currently reads: "Number of tubes plugged during the inspection outage for each active degradation mechanism."

The proposed revision reads: "Number of tubes plugged during the inspection outage for each degradation mechanism."

Item f of the Calvert Cliffs TS 5.6.9 currently reads: "Total number and percentage of tubes plugged to date."

The proposed revision reads: "The number and percentage of tubes plugged to date, and the effective plugging percentage in each steam generator,"

The licensee proposed to delete item h of the Calvert Cliffs TS 5.6.9, which currently states, "The effective plugging percentage for all plugging in each SG."

The licensee proposed to delete the word "Active" in items b and e above. Thus, all degradation mechanisms found, whether deemed to be active or not, would now be reportable. The NRC staff finds that the proposed change is acceptable, because it is more conservative.

The proposed change to combine items f and h are editorial changes that do not materially change the reporting requirements. Therefore, the NRC staff finds that these proposed changes are acceptable.

5.0 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.

6.0 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. 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 42547). 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.

7.0 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.

Principal Contributor: R. Grover

Date: October 29, 2014

DATED: October 29, 2014

AMENDMENT NO. 308 TO RENEWED FACILITY OPERATING LICENSE NO. DPR-53
CALVERT CLIFFS UNIT 1

AMENDMENT NO. 286 TO RENEWED FACILITY OPERATING LICENSE NO. DPR-69
CALVERT CLIFFS UNIT 2

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RidsNrrLAKGoldstein
RidsNrrDssStsb
RidsRgn1MailCenter
RidsAcrsAcnwMail_CTR
RGrover, NRR

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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/

Nadiyah S. Morgan, 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. 308 to DPR-53
2. Amendment No. 286 to DPR-69
3. Safety Evaluation

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ADAMS Accession No. ML14288A102 ***See dated memo**

OFFICE	LPLI-1/PM	LPLI-1/PM	LPLI-1/LA	STSB/BC	OGC	LPLI-1/BC
NAME	AChereskin	NMorgan	KGoldstein	RElliot*	JLindell	BBeasley
DATE	10/16/2014	10/17/2014	10/17/2014	10/14/2014	10/22/2014	10/29/2014

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