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U. S. Nuclear Regulatory Commission
Washington, DC 20555

ATTENTION: Document Control Desk

SUBJECT: Calvert Cliffs Nuclear Power Plant
Unit Nos. 1 & 2; Docket Nos. 50-317 & 50-318
License Amendment Request: Extension of Instrument Surveillance Intervals

REFERENCE: (a) NRC Generic Letter 91-04, Changes in Technical Specification Surveillance Intervals to Accommodate a 24-Month Fuel Cycle, dated April 2, 1991

Pursuant to 10 CFR 50.90, the Baltimore Gas and Electric Company hereby requests an Amendment to Operating Licenses Nos. DPR-53 and DPR-69 by the incorporation of the changes described below into the Technical Specifications for Calvert Cliffs Unit Nos. 1 and 2.

An Index for this submittal is provided as Attachment (3).

DESCRIPTION

The proposed amendment would revise the Calvert Cliffs Nuclear Power Plant Units 1 and 2 Technical Specifications, extending certain 18-month frequency surveillances to a refueling interval (nominally 24 months, not to exceed 30 months). Systems and equipment affected are the Reactor Protective System (RPS), Engineered Safety Features Actuation System (ESFAS), Power-Operated Relief Valve (PORV) actuation instruments, Low Temperature Overpressure Protection (LTOP)-related instruments, Remote Shutdown Panel instruments, Post-Accident Monitoring (PAM) instruments, Containment Sump Level instruments, and Radiation Monitoring instruments.

This amendment request would extend the nominal surveillance interval requirement from 18 months to a refueling interval (nominally 24 months, not to exceed 30 months) for instrument channel calibrations, RPS and ESFAS total bypass function operability verification, RPS and ESFAS time response tests, ESFAS

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Manual Trip Button channel functional tests, and ESFAS Automatic Actuation Logic Channel Functional Tests. Calvert Cliffs has been operating on a 24-month fuel cycle since July 1987 (Unit 2) and July 1988 (Unit 1), performing some Technical Specification surveillances, such as the ones described here, during mid-cycle outages. This request is the last of a series of proposed license amendments that would eliminate the need for planned mid-cycle outages to perform required surveillances.

BACKGROUND

The Calvert Cliffs Technical Specifications require certain surveillances be performed every 18 months on specified RPS instruments, ESFAS instruments, PORV-related instruments, LTOP-related instruments, Remote Shutdown Panel instruments, PAM instruments, Containment Sump level instruments, and Radiation Monitoring instruments. The 18-month interval allowed the surveillances to be performed during refueling outages for 18-month fuel cycles. The systems and their related instruments are described below. Generic Letter (GL) 91-04 (Reference a) provides guidance on issues to address when submitting requests to extend 18-month surveillance intervals to 24 months. The safety analysis done for each of the instruments used GL 91-04 as a guide in analyzing the effect on plant safety of increasing the surveillance intervals from 18 months to a refueling interval (nominally 24 months, not to exceed 30 months).

A. RPS Instruments (Technical Specification Table 4.3-1)

Reactor Protective System instruments addressed by this submittal generate a trip signal using a two-of-four logic matrix. The trip signals and associated instruments affected by this submittal are:

1. Power Level - High, ΔT Power (Table Item 2.b)

The thermal reactor power signal is generated by the ΔT power calculator contained within the Thermal Margin/Low Pressure (TM/LP) calculator circuitry, using signals from four hot leg and four cold leg temperature instruments in each of the two Reactor Coolant System (RCS) loops. Four linear power range nuclear instrumentation channels monitor reactor flux (nuclear power). The larger of either nuclear power or thermal power signal is supplied to the high power bistable trip unit, which compares the signal to the variable high power trip limit. The trip limit is determined by the variable high power trip calculator. This trip provides protection against reactivity excursions which are too rapid to be protected against by a pressurizer pressure high or thermal margin/low pressure trip.

2. Reactor Coolant Flow - Low (Table Item 3)

Reactor coolant flow rate is determined using the differential pressure of the RCS across the steam generators. The pressure signal is generated by four RCS pressure transmitters which monitor the pressure drop across each of the steam generators. The trip protects the plant from departure from nucleate boiling in the event of a sudden significant decrease in reactor coolant flow.

3. Pressurizer Pressure - High (Table Item 4)

The pressurizer pressure signal is generated by four pressurizer pressure transmitters. The Pressurizer Pressure - High trip, backed up by the pressurizer code safety valves and main steam line safety valves, provides protection against RCS overpressurization.

4. Containment Pressure - High (Table Item 5)

The containment pressure signal is generated by four containment pressure transmitters. The Containment Pressure - High trip provides assurance that a reactor trip is initiated prior to, or at least concurrent with, a safety injection actuation or high containment pressure.

5. Steam Generator Pressure - Low (Table Item 6)

The steam generator pressure signal is generated by four pressure transmitters per steam generator. The Steam Generator Pressure - Low trip provides protection against an excessive rate of heat extraction from the steam generators and subsequent cooldown of the reactor coolant.

6. Steam Generator Water Level - Low (Table Item 7)

The steam generator level signal is generated by four level transmitters per steam generator. The Steam Generator Water Level - Low trip provides core protection by preventing operation with the steam generator water level below the minimum volume required for adequate heat removal capacity and assures that the pressure of the RCS will not exceed its safety limit due to overheating. The specified setpoint in combination with the Auxiliary Feedwater Actuation System (AFAS) ensures that sufficient water inventory exists in both steam generators to remove decay heat following a loss of Main Feedwater flow event.

7. Axial Flux Offset (Table Item 8)

The Axial Flux Offset is determined using signals from the four power range excore detectors. The trip provides protection against excessive axial peaking so that it will not cause fuel damage.

8. TM/LP (Table Item 9.a)

The TM/LP trip calculator receives input signals from the power range nuclear instrumentation channels, hot leg and cold leg temperature instrumentation, Axial Flux Offset (APD) Calculator, and the ΔT Power Calculator. Reactor Coolant System pressure is input to the bistable trip unit where it is compared to the calculated TM/LP trip setpoint value or a fixed lower limit. The TM/LP trip is provided to protect against transients that could cause departure from nucleate boiling ratio to be less than a specific value.

9. Steam Generator Pressure Difference - High (Asymmetric Steam Generator Transient [ASGT] Protection Trip Function) (Table Item 9.b)

The steam generator pressure signal is generated by four steam generator pressure transmitters per steam generator. The ASGT calculator calculates the differential pressure between generators and compares it to the setpoint. A trip signal is processed by modifying the TM/LP setpoint. The High Steam Generator Pressure Difference trip provides protection against asymmetric steam generator events. The trip is designed for Anticipated Operational Occurrences associated with secondary system malfunctions which result in asymmetric primary loop coolant temperatures.

10. Wide Range Logarithmic Neutron Flux Monitor Signal Rate of Change of Power - High (Table Item 11)

The Wide Range Logarithmic Neutron Flux Monitor trip protects the core during startup operations and its use serves as a backup to the administratively-controlled startup rate limit. Four wide range logarithmic neutron monitoring channels provide input signals to the RPS where the rate of change of power is compared to a fixed trip setpoint value. The startup rate trip is not credited in any Updated Final Safety Analysis Report (UFSAR) accident analyses.

B. RPS Instrument Total Bypass Functions (Technical Specification 4.3.1.1.2)

Manually or automatically inserted bypasses allow plant operation under conditions that do not require the respective RPS functions, but would result in an unnecessary RPS trip if they were not bypassed. The total bypass functions remove these bypasses when the plant conditions no longer support allowing the bypasses. The RPS total bypass functions are accomplished by three bistables: the zero power mode bypass bistable (wide range nuclear instrument channels), the low steam generator pressure trip bypass bistable, and the 15% power level bypass bistable (linear power range nuclear instrument channel).

1. Zero Power Mode Bypass for Reactor Coolant Flow - Low, TM/LP and Steam Generator Differential Pressure Trips

The Reactor Coolant Low Flow, TM/LP, and Steam Generator Differential Pressure trips may be bypassed below $10^{-4}\%$ of rated power and the bypasses are automatically removed (e.g., the trips are enabled) when power is greater than or equal to $10^{-4}\%$ of rated power. The removal is accomplished by means of the wide range nuclear instrument $10^{-4}\%$ bistable. The reactor power signal is generated by four wide range nuclear instruments.

2. Low Steam Generator Pressure Trip Bypass

The Steam Generator Low Pressure Trip may be bypassed below 785 psia and the bypass is automatically removed at or above 785 psia. The removal is accomplished by means of the Steam Generator Low Pressure Trip Bypass bistable. The steam generator pressure signal is generated by four steam generator pressure transmitters per steam generator.

3. Axial Flux Tilt Trip Bypass

The Axial Flux Tilt Trip is automatically bypassed when reactor power decreases below 15% of rated power, as per the Technical Specifications, and the bypass is automatically removed when reactor power rises above 15% of rated power. The removal is accomplished by means of the linear range channel 15% bistable. The reactor power signal is generated by four linear power range reactor power detectors.

4. Wide Range Logarithmic Neutron Flux Monitor Signal Rate of Change of Power - High Bypass

Technical Specifications allow the rate of Rate of Change of Power Trip to be bypassed below 10^{-4} % and above 12% of rated power. This is accomplished using the wide range nuclear instrument 10^{-4} % bistable and linear range 15% bistables. The 10^{-4} % of rated power signal is generated by four wide range nuclear instrument channels, and the 15% of rated power signal is generated by four linear power range instrument channels.

C. RPS Reactor Trip Response Time (Technical Specification 4.3.1.1.3)

The RPS reactor trip response time is tested to verify that the reactor trip breakers open within the time interval assumed in the accident analyses.

D. ESFAS instruments (Technical Specification Table 4.3-2).

Engineered Safety Features Actuation System instruments addressed by this submittal generate an actuation signal using a two-of-four logic matrix. The ESFAS output signals pass through the Automatic Actuation Circuitry. The ESFAS signals and associated instruments affected by this submittal are:

1. Safety Injection Actuation Signal (SIAS) (Table Item 1.b and c, Table Notation Item 2 and 3)

The SIAS actuates equipment necessary for cooling and reactivity control following a loss-of-coolant accident (LOCA) and other design basis events. The SIAS is initiated by a Pressurizer Pressure - Low signal or by a Containment Pressure - High signal. These signals are generated by four pressurizer pressure transmitters and four containment pressure transmitters, respectively.

2. Containment Spray Actuation Signal (CSAS) (Table Item 2.b, Table Notation Item 6)

The CSAS actuates equipment necessary in the event of a LOCA and other design basis events. The CSAS is initiated by a Containment Pressure - High signal. This signal is generated by four containment pressure transmitters.

3. Containment Isolation Signal (CIS) (Table Item 3.b, Table Notation Item 4)

The CIS actuates equipment to isolate the containment from the outside environment. The CIS is initiated by a Containment Pressure - High signal. The signal is generated by four containment pressure transmitters.

4. Main Steam Line Isolation (Table Item 4.b, Table Notation Item 5)

The steam generator isolation signal (SGIS) shuts the Main Steam and Main Feedwater Isolation Valves and stops the Main Feedwater System pumps in the event of an excessive loss of steam from the Main Steam System. The SGIS is initiated by a Steam Generator Pressure - Low signal. The signal is generated by four steam generator pressure transmitters per steam generator.

5. Recirculation Actuation Signal (RAS) (Table Item 5.b)

The RAS secures the injection of water into the RCS from the Refueling Water Tank (RWT) initiated by a SIAS, and actuates equipment to begin recirculation of the water in the containment for long-term core cooling. The RAS is initiated by an RWT - Low level signal, indicating that sufficient water for long-term cooling has been injected into the containment. The signal is generated by four RWT level switches.

6. Containment Purge Valve Isolation Signal (Containment Radiation Signal (CRS)) (Table Item 6.b)

When the containment atmosphere is being purged through the two in-line containment purge exhaust valves, the CRS actuates equipment to prevent the release of radioactive material to the environment in the event of a reactor coolant leak, a shielding failure, or a fuel pin failure when the reactor vessel head is removed. The signal also provides audible and visual warning of high radiation levels in the containment. The signal is generated by four radiation detectors in the containment.

7. Loss Of Power - 4.16 kV Bus Undervoltage Signal (Table Item 7.a and .b)

The 4.16 kV bus undervoltage signal initiates the Emergency Diesel Generator (EDG) start and load sequencing signals automatically to provide reliable emergency power from the EDGs for loads necessary to shut down the plant safely and maintain it in a safe shutdown condition. The two sets of relays that can initiate the undervoltage signal are the degraded grid and loss of voltage relays. The degraded grid relay initiates the signal in the event that an undervoltage condition is sustained on the 4.16 kV busses for more than eight seconds. The loss of voltage relay initiates a signal in the event extremely low voltage is sustained on the 4.16 kV busses for more than two seconds. The 4.16 kV bus voltage is stepped down by potential transformers which provide a voltage signal to redundant undervoltage relays. Each 4.16 kV bus has four degraded grid relays and four loss of voltage relays.

8. Chemical and Volume Control System Isolation Signal (CVCIS) (Table Item 8)

The CVCIS isolates the letdown portion of the Chemical and Volume Control System (CVCS) from the RCS in the event of a CVCS leak. The CVCS consists of components that control RCS coolant volume and chemistry, reduce coolant radioactivity, and provide reactivity control by controlling the boron concentration of the primary coolant. Most of the components are located outside the containment in the West Penetration Room and the Letdown Heat Exchanger Room. Two letdown stop valves inside containment can isolate the letdown portion of the CVCS from the RCS. The CVCIS prevents the consequences of a CVCS line break outside the containment by shutting the letdown isolation valves if high room pressure is detected in the West Penetration Room or the Letdown Heat Exchanger Room. The CVCIS is generated by four pressure transmitters, two in the West Penetration Room and two in the Letdown Heat Exchanger Room.

9. Auxiliary Feedwater (AFW) (Table Item 9.b and .c)

The AFAS - Start signal causes one steam and one electric AFW System pump to start which will provide feedwater to the steam generators to maintain steam generator level. Auxiliary Feedwater Actuation System - Start is initiated by a Steam Generator Level - Low signal. This signal is generated by four steam generator level transmitters per steam generator.

The AFAS - Block signal causes redundant AFW System block valves to close, stopping AFW flow to a faulted steam generator. Auxiliary Feedwater Actuation System - Block is initiated by a Steam Generator Differential Pressure - High signal. This signal is generated by four steam generator pressure transmitters per steam generator.

E. ESFAS Manual Trip Buttons (Technical Specification Table 4.3-2, Items 1a, 2a, 3a, 4a, 5a, 6a, and 9a)

The SIAS, CSAS, CIS, RAS, Containment Purge Isolation Valves, and AFAS trip systems have manual trip buttons, and the SGIS has Main Steam Isolation Valve and Feedwater Header Isolation hand switches. These buttons and handswitches are a backup means for operators to initiate an ESFAS actuation if necessary.

F. ESFAS Instrument Total Bypass Functions (Technical Specification 4.3.2.1.2)

Manually inserted bypasses allow plant evolutions under conditions that do not require the respective ESFAS functions, but would result in an ESFAS actuation if they were not bypassed. The total bypass functions remove the bypasses when the plant conditions no longer warrant the bypass.

1. SIAS Block, Pressurizer Pressure - Low

The Pressurizer Pressure Bypass allows the operator to bypass the SIAS and prevent safeguards systems from actuating during normal depressurization of the RCS. The total

bypass logic removes the bypass automatically above a preset pressure. The pressurizer pressure signal is generated by four pressurizer pressure transmitters.

2. SGIS Block, Steam Generator Pressure - Low

The Steam Generator Pressure Bypass function allows the operator to bypass the SGIS and prevent safeguards systems from actuating during normal depressurization of the steam generators. The total bypass function automatically removes the bypass above a preset pressure. The steam generator pressure signal is generated by four steam generator pressure transmitters per steam generator.

G. Engineered Safety Features Response Time (Technical Specification 4.3.2.1.3)

The Engineered Safety Features response time is measured to verify that equipment actuates within the time interval assumed in the accident analyses.

H. PORVs (Technical Specification 4.4.3.1.b)

Power-Operated Relief Valves are set to lift before the pressurizer code safety valves and subsequently reseal to minimize the release of reactor coolant from the RCS. The RPS Pressurizer Pressure - High trip signal initiates the signal to open the PORVs. The Pressurizer Pressure - High RPS trip, backed up by the pressurizer code safety valves and main steam line safety valves, provides RCS protection against overpressurization.

I. Low Temperature Overpressure System (Technical Specification 4.4.9.3.1.b.)

The Low Temperature Overpressure System provides protection against RCS overpressurization at low temperature by a combination of administrative controls and hardware. The hardware includes two PORVs with variable pressurizer pressure setpoints when operating in the LTOP region. Each PORV protection circuit is provided input by one pressurizer pressure transmitter and one RCS loop cold leg temperature detector dedicated to the specific circuits.

J. Remote Shutdown Panel Indication (Technical Specification Table 4.3-6)

Plant parameter indications are provided to operators on a Remote Shutdown Panel to be used while placing and maintaining the plant in a safe shutdown condition in the event the Control Room is uninhabitable. The indications are used to verify proper system response to plant conditions and operator actions. The Remote Shutdown Panel instruments addressed by this submittal are:

1. Reactor Coolant Cold Leg Temperature (Wide Range) (Table Item 3)

Two temperature sensors per RCS loop provide the input to the Remote Shutdown Panel indication.

2. Pressurizer Pressure (Wide Range) (Table Item 4)

Two pressurizer pressure transmitters provide input to the Remote Shutdown Panel indication.

3. Pressurizer Level (Table Item 5)

Two pressurizer level transmitters provide input to the Remote Shutdown Panel indication.

4. Steam Generator Level (Wide Range) (Table Item 6)

Two level transmitters per steam generator provide input to the Remote Shutdown Panel indication.

5. Steam Generator Pressure (Table Item 7)

Two pressure transmitters per steam generator provide input to the Remote Shutdown Panel indication.

K. PAM System (Technical Specification Table 4.3-10)

The PAM instruments provide the Control Room operators with primary information necessary to take manual actions, as necessary, in response to design basis events, and to verify proper system response to plant conditions and operator actions. The PAM instruments addressed by this submittal are:

1. Containment Pressure (Table Item 1)

Three containment pressure transmitters with overlapping ranges provide input to the Control Room indication.

2. Reactor Coolant Outlet Temperature (Wide Range) (Table Item 3)

One reactor coolant outlet temperature sensor per primary coolant loop provides input to the Control Room indication.

3. Pressurizer Pressure (Wide Range) (Table Item 4)

Two pressurizer pressure transmitters provide the input to the Control Room indication.

4. Pressurizer Level (Wide Range) (Table Item 5)

Two pressurizer level transmitters provide the input to the Control Room indication.

5. Steam Generator Pressure (Table Item 6)

Four steam generator pressure transmitters per steam generator provide the input to the Control Room indication.

6. Steam Generator Level (Wide Range) (Table Item 7)

Four steam generator level transmitters per steam generator are provided. Two provide the input to the Control Room indication and two provide input to the Remote Shutdown Panel indication.

7. AFW Flow Rate (Table Item 8)

There is one AFW line per steam generator, and two flow rate detectors per AFW line which provide the input to the Control Room indication.

8. RCS Subcooled Margin Monitor (Table Item 9)

The Subcooled Margin Monitor calculates the RCS margin to saturation based on the pressure/temperature inputs. Two Subcooled Margin Monitors provide input to the Control Room indication.

9. PORV/Safety Valve Acoustic Monitors (Table Item 10)

The acoustic monitors for the Pressurizer PORV and Safety Valves provide one of the indications that the PORVs or Pressurizer Safety Valves have lifted and reactor coolant is being released from the RCS. Flow is detected by measuring the vibration of the piping through which the coolant is released. Two detectors sense coolant flow from the PORVs, and two sense coolant flow from the Safety Valves, and all four channels provide input to the Control Room indication.

10. Feedwater Flow (Table Item 12)

One flow rate transmitter in each Main Feedwater line provides input to the Control Room indication.

11. Containment Water Level (Wide Range) (Table Item 13)

Containment water level provides one of the indications that a LOCA has occurred. Two level transmitters provide input to the Control Room indication.

L. Containment Sump Level Alarm System (Technical Specification 4.4.6.1.b.)

The Calvert Cliffs containment has a drainage sump located in the containment basement. The Containment Sump Level Alarm System provides an alarm in the Control Room to provide one of the available indications of excessive RCS leakage during normal plant operation. The sump

collects drain water from components in the containment, the reactor cavity cooling plenum, and the reactor cavity. Two redundant level switches provide the signal for the high level alarm.

M. Radiation Monitoring Instrumentation (Technical Specification Table 4.3-3.1)

Containment Radiation Monitoring Systems actuate equipment and/or provide an alarm in the event of high radiation conditions inside the containment. The Radiation Monitoring instruments addressed by this submittal are:

1. Containment Purge and Exhaust Isolation Radiation Monitoring System (Table Item 1.a)

The Containment Purge System is used to ensure a suitable environment for personnel in the containment when the reactor is in cold shutdown or is being refueled. When the containment atmosphere is being purged, the Containment Radiation Signal (CRS) actuates equipment to prevent the release of radioactive material to the environment. The CRS is actuated by high radiation levels in containment from events such as a reactor coolant leak, a shielding failure, or a fuel pin failure when the reactor vessel head is removed. The signal also provides audible and visual warning. The signal is generated by four radiation detectors in containment.

2. Containment Area High Range Radiation Monitoring System (Table Item 1.b)

The Containment Area High Range Radiation Monitoring System consists of two redundant gamma detectors located inside the containment with readout and alarm modules in the Control Room. The system provides an indication of high radiation levels in containment. This is one of the indications available to operators of a large LOCA. The radiation alarm setpoints are set to alert operators to a severe accident without causing excessive spurious alarms.

REQUESTED CHANGE

Revise the Calvert Cliffs Units 1 and 2 Technical Specifications 4.3.1.1.2, 4.3.1.1.3, Table 4.3-1, 4.3.2.1.2, 4.3.2.1.3, Table 4.3-2, Table 4.3-3, Table 4.3-6, Table 4.3-10, 4.4.3.1.b, 4.4.6.1.b, and 4.4.9.3.1.b, as shown on the marked-up pages attached to this transmittal. These changes will extend the surveillance interval from 18 months to a refueling interval (nominally 24 months, not to exceed 30 months).

SAFETY ANALYSIS

Generic Letter 91-04 describes issues licensees should evaluate when extending 18-month surveillances to 24 months to accommodate 24-month fuel cycles. The Generic Letter provides guidance on how licensees should evaluate the effect on safety and the evaluation should support a conclusion that the effect on safety is small.

The analyses of the instrument functions addressed in this submittal have certain generic attributes. To explain the methodology used in analyzing the instrument functions, and provide some of the technical data, these generic attributes are discussed in the Generic Safety Analysis Issues section. The subsequent safety analyses for each of the instrument functions provide the discussions regarding the effect of this surveillance interval extension request.

A. GL 91-04 Issues

Specific GL 91-04 issues are addressed in the analyses for instruments that are subject to drift and provide input for automatic actuations. In cases where the instruments are not subject to drift or don't provide input for automatic actuations, alternate discussions are provided. The GL 91-04 issues are:

- Issue 1. Confirm that instrument drift as determined by as-found and as-left data from surveillance and maintenance records has not, except on rare occasions, exceeded acceptable limits for a calibration interval.
- Issue 2. Confirm that the values of drift for each instrument type (make, model, and range) and application have been determined with a high probability and a high degree of confidence. Provide a summary of the methodology and assumptions used to determine the rate of instrument drift with time based upon historical plant calibration data.
- Issue 3. Confirm that the magnitude of instrument drift has been determined with a high probability and a high degree of confidence for a bounding calibration interval of 30 months for each instrument type (make, model, number, and range) and application that performs a safety function. Provide a list of the channels by Technical Specification section that identifies these instrument applications.
- Issue 4. Confirm that a comparison of the proposed instrument drift errors has been made with the values of drift used in the setpoint analysis. If this results in revised setpoints to accommodate larger drift errors, provide proposed Technical Specification changes to update trip setpoints. If the drift errors result in a revised safety analysis to support existing setpoints, provide a summary of the updated analysis conclusions to confirm that safety limits and safety analysis assumptions are not exceeded.
- Issue 5. Confirm that the projected instrument errors caused by drift are acceptable for control of plant parameters to effect a safe shutdown with the associated instrumentation.
- Issue 6. Confirm that all conditions and assumptions of the setpoint and safety analyses have been checked and are appropriately reflected in the acceptance criteria of plant surveillance procedures for channel checks, channel functional tests, and channel calibrations.
- Issue 7. Provide a summary description of the program for monitoring and assessing the effects of increased calibration surveillance intervals on instrument drift and its effect on safety.

B. Generic Safety Analysis Issues

1. Rosemount Transmitters

This discussion primarily addresses GL 91-04 Issues 1, 2 and 3. Many of the instruments addressed in this license amendment request will have or have had a new brand of transmitter installed, Rosemount Models 1152, 1153 or 1154. This replacement is to improve instrument reliability and performance, and reduce instrument drift. In many cases, this was done specifically to improve instrument performance when evaluating the effects of the increased surveillance intervals, rather than justifying the performance of existing instruments. Calvert Cliffs plant-specific as-found and as-left data were not used to make a determination of drift for newly installed Rosemount transmitters because there was limited or no plant data available for statistical analysis. However, the manufacturer testing demonstrated that the Rosemount sensor parameters are bounded by a 30-month stability specification of $\pm 0.2\%$ of the upper range limit (URL).

2. WEED Resistance Temperature Detectors (RTDs)

This discussion primarily addresses GL 91-04 Issues 1, 2 and 3. The temperature sensors used by the instruments addressed in this license amendment request are WEED brand RTDs. WEED brand RTDs were installed a few years ago. Instrument drift of nuclear grade RTDs has been demonstrated and quantified by industry testing. Calvert Cliffs plant-specific as-found and as-left data were not used to make a determination of drift because there were limited as-found and as-left data available for statistical analysis. However, industry testing (NUREG/CR-5560, EPRI/RP-2409-15) has demonstrated that nuclear grade RTDs are bounded by a 30-month drift specification of $\pm 0.36^{\circ}\text{F}$.

3. Channel Functional Testing

Instrument loop surveillances for the RPS, ESFAS, PORV and LTOP functions are split into two tests, sensor testing and balance of loop testing. Both sensor and balance of loop components are fully calibrated at each refueling. A channel functional test is performed on the balance of loop components more frequently than on a refueling interval basis. Channel functional tests results would indicate potential calibration problems. Channel calibrations on the balance of loop components would be performed during plant operation if deemed appropriate based on the channel functional test results. Thus, the balance of loop components are not affected by this surveillance interval extension request. For this reason, a detailed drift analysis and maintenance review was not done on the balance of the instrument loops for these functions.

4. Safe Plant Shutdown

In addressing GL 91-04 Issue 5, automatic equipment response and operator indications were considered when evaluating the ability to effect a safe shutdown with the associated instrumentation. The 30-month (24 months + 25%) uncertainties calculated for instrument functions that automatically actuate equipment were found to be bounded by the

uncertainties assumed in the safety analyses. The primary guidance which operators use to control the plant in response to an accident or transient to effect a safe shutdown are the Emergency Operating Procedures (EOPs). The Safety Function Status Checklists contained in the EOPs are used to maintain the primary plant parameters within limits so safety functions are met to effect a safe shutdown. Generally, the PAM and/or the Remote Shutdown instruments designated in the Technical Specifications are used to confirm the EOP safety functions are being met. Calculated uncertainties for these instruments were determined to be acceptable for control of plant parameters to effect a safe shutdown.

5. Instrument Performance Monitoring

In addressing GL 91-04 Issue 7, instrument performance monitoring will be done between refueling interval (nominally 24 months, not to exceed 30 months) channel calibrations by methods that include channel checks and/or channel functional tests, or routine monitoring. Channel checks, channel functional tests, and routine monitoring of indications provide a reliable indication of instrument operation. These methods have identified improperly operating instruments in the past. The as-found calibration data collected during refueling outages are evaluated as part of the surveillance program. The plant corrective action program is used when instrument parameters fall outside specified acceptance criteria. An evaluation was performed to verify that instrument parameter bands used in the surveillances would identify degrading instrument performance. We expect the instrument performance monitoring program described above to identify, and initiate appropriate action in response to, problems associated with drift that could potentially cause plant parameters to exceed accident analysis assumptions.

6. Setpoints

In addressing GL 91-04 Issues 4 and 6, Total Loop Uncertainties for 30-month surveillance intervals were used to evaluate actuation setpoints and their relation to analytical limits for the RPS, ESFAS, PORV, LTOP, Containment Sump Level and Radiation Monitoring instruments, and no case was found where the setpoints needed to be revised. The Remote Shutdown and PAM instruments have no setpoints for automatic equipment response.

7. Drift Analysis, Instrument Uncertainty and Setpoint Methodology

Instrument loops affected by this request were subjected to a detailed engineering review to establish the basis for a 30-month (24 months +25%) calibration frequency. As requested by GL 91-04, drift analyses were completed as needed. These analyses were performed using the Calvert Cliffs calculation procedure. The drift analyses incorporate the guidance provided by EPRI Document TR-103335, March 1994 "Guidelines for Instrument Calibration Extension/Reduction Programs," ISA-dRP67.04, Part II, Draft Recommended Practice, "Methodologies for the Determination of Setpoints for Nuclear Safety-Related Instrumentation," Draft 10, and ISA-S-67.04-1987, "Standard for Nuclear Safety-Related Instrumentation."

8. Time Response Testing

Time response testing is performed to ensure that required automatic RPS and ESFAS instrumentation-related actions are carried out within time limits assumed in accident analyses. There is no technical basis for expecting the time response of the ESFAS or RPS trip channels to be time dependent. Nevertheless, time response data collected since 1976 were evaluated for several functions. The functions chosen were those with the shortest allowable response times. These data displayed the expected random positive and negative changes in the measured time response, confirming the lack of time dependency. Still, a 10-year drift value was calculated using conservative assumptions to assess the likelihood of exceeding the UFSAR time response limit over a 10-year period. The response times were found to remain within acceptable limits.

9. Nuclear Instrument Calibration

Reactor Protective System wide range and power range nuclear instrument neutron detectors are excluded from refueling interval channel calibration, in accordance with the Technical Specifications. The power range nuclear instrument channels are calibrated daily to the calorimetric calculation. A channel functional test of the wide range nuclear instrument channel is performed prior to each plant startup. Therefore, the nuclear instrument channels are not adversely affected by this surveillance interval extension.

10. Effect on Plant Safety

Our conclusion that the surveillance interval extensions effect on plant safety is small and would not invalidate any assumption in the plant licensing basis is based on the extension not requiring any setpoint changes and plant parameter indications still being acceptable for control of plant parameters to effect a safe shutdown.

C. RPS, ESFAS, PORV and LTOP Instruments

The approach used to evaluate the RPS, ESFAS, PORV and LTOP instruments was similar for most instrument functions. Exceptions are the RPS Wide Range Logarithmic Neutron Flux Monitor Signal Rate of Change of Power, RAS, Containment Purge Isolation Signal, and the Loss of Power 4.16 kV bus undervoltage signal, which are discussed separately. Also discussed in this section are the RPS Total Bypass Logic Functions, RPS Time Response Testing, ESFAS Automatic Actuation Logic Circuitry, ESFAS Total Bypass Logic Functions, ESFAS Time Response Testing, and ESFAS Manual Trip Pushbuttons and Handswitches.

1. Generic RPS, ESFAS, PORV and LTOP Discussion

The RPS, ESFAS, PORV and LTOP functions addressed in this section are:

RPS Trip Function (Technical Specification Table 4.3-1)	
High Thermal Power (Table Item 2.b)	
Low RCS Flow (Table Item 3)	
High Pressurizer Pressure (Table Item 4)	
High Containment Pressure (Table Item 5)	
Low Steam Generator Pressure (Table Item 6)	
Low Steam Generator Water Level (Table Item 7)	
Axial Flux Offset (Table Item 8)	
TM/LP (Table Item 9.a)	
High Steam Generator Pressure Difference (Table Item 9.b)	

ESFAS Actuation (Technical Specification Table 4.3-2)	
High Containment Pressure SIAS (Table Item 1.b)	
Low Pressurizer Pressure SIAS (Table Item 1.c)	
High Containment Pressure CSAS (Table Item 2.b)	
High Containment Pressure CIS (Table Item 3.b)	
Low Steam Generator Pressure SGIS (Table Item 4.b)	
High West Penetration Room and Letdown Heat Exchanger Room Pressure CVCIS (Table Item 8)	
Low Steam Generator Level AFAS start (Table Item 9.b)	
High Steam Generator Pressure Difference AFAS Block (Table Item 9.c)	
High Pressurizer Pressure PORV Opening (Technical Specification 4.4.3.1.b)	
High Pressurizer Pressure	
LTOP PORV Opening (Technical Specification 4.4.9.3.1.b)	
Pressurizer Pressure	
RCS Cold Leg Temperature	

- Issue 1. Resistance Temperature Detector, pressure transmitter, and level transmitter drift have been quantified as described in the Generic Safety Analysis Issues section for Rosemount transmitters and nuclear grade RTDs. Channel functional testing on the balance of the instrument loops is done more frequently than the calibration of the sensors, and is not affected by this surveillance interval extension request. Problems requiring maintenance have been primarily identified by non-calibration-related surveillances and observations.
- Issue 2. Methodologies used to analyze drift are discussed in the Generic Safety Analysis Issues section. Reactor Protective System, ESFAS, PORV and LTOP-related pressure and level transmitters evaluated were produced by Rosemount. Analysis for Rosemount transmitters was based on

manufacturer testing data described in the Generic Safety Analysis Issues section. The RPS and LTOP-related RTDs were made by WEED, and analysis was based on industry testing data as described in the Generic Safety Analysis Issues section. The balance of the instrument loops are not affected by this surveillance interval extension request because their channel functional testing is done more frequently than the sensor calibration.

- Issue 3. Reactor Protective System, ESFAS, PORV and LTOP-related pressure and level transmitters evaluated were produced by Rosemount. Analysis for Rosemount transmitters was based on manufacturer testing data described in the Generic Safety Analysis Issues section. The RPS and LTOP-related RTDs were made by WEED, and analysis was based on industry testing data as described in the Generic Safety Analysis Issues section. The balance of the instrument loops are not affected by this surveillance interval extension request because their channel functional testing is done more frequently than the sensor calibration.
- Issue 4. Instrument and setpoint uncertainty calculations have been revised as required to reflect the extended surveillance interval drift. Comparison of these uncertainties with the assumptions made in the Setpoint Analyses found that the current setpoints are unaffected by the extended surveillance interval drift, and no setpoints required changing.
- Issue 5. These automatic protective actions support safe shutdown. Instrument and uncertainty calculations have been revised to reflect the extended surveillance interval drift. These calculations demonstrate that the current setpoints are bounding and that the protective functions are unaffected by the extended surveillance interval.
- Issue 6. Instrument and setpoint uncertainty calculations have been revised to reflect the extended surveillance interval drift. These calculations demonstrate that the current setpoints are unaffected by the extended surveillance interval. Channel check criteria and applicable surveillance calibration tolerances provide reliable criteria for detection of degraded channel operation. The relations of the actuation setpoints to the analytical limits were evaluated, and no cases were found where the setpoints needed to be revised. A review of station surveillance and calibration procedures indicates that the established setpoints have been properly implemented in applicable surveillance tests.
- Issue 7. Instrument performance monitoring will be done between refueling interval (nominally 24 months, not to exceed 30 months) channel calibrations by channel checks and/or channel functional tests, and routine monitoring. These methods provide a reliable indication of instrument operation and have identified improperly operating instruments in the past.

As found calibration data is evaluated as part of the channel calibration surveillance program. The plant corrective action program is used when instrument parameters fall outside specified acceptance criteria. An evaluation was performed to verify that instrument parameter bands used in the surveillances would identify degrading instrument performance. We expect the instrument performance monitoring program described above to identify and initiate action in response to problems associated with drift that would potentially cause plant parameters to exceed accident analysis assumptions.

2. RPS Wide Range Logarithmic Neutron Flux Monitor Trip (Table 4.3-1, Item 11)

The Wide Range Logarithmic Neutron Flux Monitor Startup Rate trip receives input from the wide range neutron detectors and is not explicitly credited in UFSAR accident analyses. The trip provides a backup to the administratively-controlled startup rate limit. The neutron detectors, in accordance with the Technical Specifications, are excluded from channel calibration, and the RPS startup rate channel functional test is performed prior to reactor startup. All the surveillances are thus performed prior to reactor startup, and the instrument only serves its function during reactor startup. Therefore, the effect of this surveillance interval extension on safety is small, and would not invalidate any assumption in the plant licensing basis.

3. ESFAS (RWT Level - Low) RAS (Technical Specification 4.3.2.1.1, Table 4.3-2, Item 5.b)

The RAS actuation circuitry receives input from level switches and uses ESFAS bistable trip units. Refueling water tank level switches are mechanical devices. Their quarterly channel functional tests are equivalent to the channel calibrations. Since the level switch channel calibrations and ESFAS bistable loop channel functional tests are performed on a quarterly basis, instrument performance is unaffected by the extended surveillance interval. Therefore, the effect of the surveillance interval extension on safety is small, and would not invalidate any assumption in the plant licensing basis.

4. ESFAS (Containment Purge Valve Isolation) CRS (Technical Specification Table 4.3-2, Item 6.b)

The CRS loops are not required to achieve a safe shutdown, and are credited only in the fuel handling design basis accident. Containment Radiation Signal actuation circuitry receives input from radiation detectors and uses ESFAS bistable trip units. The CRS loops are required only in Mode 6. Prior to entering Mode 6, the loops are calibrated and functionally tested, and returned to service. Therefore, the effect of the surveillance interval extension on safety is small, and would not invalidate any assumption in the plant licensing basis.

5. ESFAS (4.16 kV Bus Undervoltage) EDG Diesel Generator Start (Technical Specification Table 4.3-2, Items 7.a and 7.b)

The 4.16 kV actuation circuitry receives input from undervoltage relays and uses ESFAS digital bistable trip units. Statistical analysis of historical as-found and as-left calibration data for the undervoltage relays and a review of historical maintenance records was performed, as described in the Generic Safety Analysis Issues section. This analysis confirmed that the undervoltage relays are unaffected by extended surveillance intervals and that the existing plant setpoints are unaffected by the extended surveillance interval drift. The ESFAS bistable loops are unaffected by the surveillance interval extension request because operability is verified quarterly via a channel functional test. All the circuitry downstream of the undervoltage relays is digital logic and not subject to drift effects, and so was not evaluated. Problems requiring maintenance have been primarily identified by non-calibration-related surveillances and observations. The loss of voltage relay setting is not a critical value since its function is to detect a complete loss of voltage. For this reason, no calculations were performed on this function. A review of station surveillance and calibration procedures indicates that the established setpoints have been properly implemented. As found calibration data are evaluated as part of the channel calibration surveillance program. Based on the factors described above, the existing plant setpoints are unaffected by the extended surveillance interval, and surveillance procedures provide assurance that failed or degraded loop components are detected so that appropriate action can be taken.

6. RPS Total Bypass Logic Functions (Technical Specification 4.3.1.1.2)

The RPS Total Bypass Logic Functions exist for the Zero Power Mode Bypass, Low Steam Generator Pressure Trip Bypass, Axial Flux Tilt Trip Bypass, and Wide Range Logarithmic Neutron Flux Monitor Signal Rate of Change of Power - High bypass. The instruments that provide input to these bypasses are the steam generator pressure transmitters, Wide Range Logarithmic Neutron Flux detectors, and the linear power range nuclear instruments. The steam generator pressure transmitter drift was previously discussed for the RPS Low Steam Generator Pressure trip. The wide range and power range nuclear instrument neutron detectors are excluded from channel calibrations during steady power operation. The Wide Range Logarithmic Neutron Flux Monitor channels are calibrated and channel functional tests are performed prior to plant startup. The balance of the instrument loops for the steam generator pressure instruments and linear range nuclear instruments are not affected by this surveillance interval request, because channel functional tests are done more frequently than channel calibrations. Thus, based on drift analysis and surveillance testing, extending the surveillance intervals from every 18 months to a refueling interval (nominally 24 months, not to exceed 30 months) will have no impact on the ability to detect degraded operation of this circuitry, and the effect on safety is small.

7. RPS Time Response Testing (Technical Specification 4.3.1.1.3)

Response time of RPS components was evaluated as described in the Generic Safety Analysis Issues section. An evaluation of the response time data and other factors affecting response time testing, as described in the Generic Safety Analysis Issues section, determined that the surveillance extension did not have a significant effect on the response times, which remained within acceptable limits. Therefore, the effect of the surveillance interval extension on safety is small, and would not invalidate any assumption in the plant licensing basis.

8. ESFAS Automatic Actuation Logic Circuitry (Technical Specification 4.3.2.1.1, Table 4.3-2, Notation 2, 3, 4, 5, and 6)

The Automatic Actuation Logic circuitry is tested quarterly, but specific SIAS, CSAS, CIS, and SGIS functions are tested during shutdown because they are exempted from testing during power operation. The circuitry is digital-based and not subject to drift mechanisms. Therefore, the effect of the surveillance interval extension on safety is small, and would not invalidate any assumption in the plant licensing basis.

9. ESFAS Total Bypass Logic Functions (Technical Specification 4.3.2.1.2)

The ESFAS Total Bypass Logic Functions exist for the SIAS Low Pressurizer Pressure and SGIS Low Steam Generator Pressure actuations. The transmitter drift for the respective instruments has been previously evaluated in this submittal. The balance of the instrument loop is not affected by this surveillance interval extension request, because channel functional testing is done more frequently than channel calibration. Thus, based on drift analysis and surveillance testing, extending the surveillance intervals from every 18 months to a refueling interval (nominally 24 months, no to exceed 30 months) will have no effect on the ability to detect degraded operation of this circuitry, and the effect on safety is small.

10. ESFAS Time Response Testing (Technical Specification 4.3.2.1.3)

Response time testing of ESFAS components was evaluated as described in the Generic Safety Analysis Issues section. An evaluation of the response time data and other factors affecting response time evaluation, as described in the Generic Safety Analysis Issues section, determined that the surveillance extension did not have a significant effect on the response times, which remained within acceptable limits. Therefore, the effect of the surveillance interval extension on safety is small, and would not invalidate any assumption in the plant licensing basis.

11. ESFAS Manual Trip Push Buttons and Handswitches (Technical Specification 4.3.2.1.1, Table 4.3-2, Items 1a, 2a, 3a, 4a, 5a, 6a, and 9a)

The SIAS, CSAS, CIS, RAS, Containment Purge Valves Isolation, and AFAS trip systems have manual trip buttons, and the SGIS has Main Steam Isolation Valve and

Feedwater Header Isolation handswitches. These buttons and handswitches are a backup means for operators to initiate an ESFAS actuation if necessary. The 18-month test checks that the push buttons actuate the respective ESFAS functions, and that the handswitches actuate their respective valves. A review of maintenance history indicates that these components have not failed when tested and there is no evidence that they will fail due to an extended surveillance interval. This is a channel functional test, not a calibration, so no drift analysis is required. Therefore, the effect of the surveillance interval extension on safety is small, and would not invalidate any assumption in the plant licensing basis.

D. Remote Shutdown Instrumentation

The primary Remote Shutdown instrument loop functions evaluated were:

Remote Shutdown Indication (Technical Specification Table 4.3-6)
Reactor Coolant Cold Leg Temperature (Table Item 3)
Wide Range Pressurizer Pressure (Table Item 4)
Uncompensated Pressurizer Level (Table Item 5)
Wide Range Steam Generator Level (Table Item 6)
Steam Generator Pressure (Table Item 7)

For each of the Remote Shutdown instruments addressed, the calculated drift was determined to be acceptable for control of plant parameters to effect a safe shutdown. There are no Remote Shutdown Instrumentation setpoints associated with any automatic functions. The Remote Shutdown pressure and level transmitters evaluated were produced by Rosemount. Analysis for Rosemount transmitters was based on manufacturer testing data described in the Generic Safety Analysis Issues section. The analysis of the Remote Shutdown-related RTDs was based on industry testing data as described in the Generic Safety Analysis Issues section. The methodology used to analyze drift is discussed in the Generic Safety Analysis Issues section. An evaluation of maintenance history indicates that the indication loops have operated satisfactorily over their lifetime and no adverse trends in operability have been identified. Problems requiring maintenance have been primarily identified by channel checks and observations. Monthly channel check criteria and applicable surveillance calibration tolerances incorporating the drift analyses were established to continue to provide reliable detection of degraded channel operation. As-found calibration data are evaluated as part of the channel calibration surveillance program. Therefore, surveillance procedures provide assurance that failed or degraded loop components are detected so that appropriate action can be taken.

The projected 30-month uncertainties caused by drift were evaluated to be acceptable for control of the plant to effect a safe shutdown. Therefore, the effect of the surveillance interval extension on safety is small, and would not invalidate any assumption in the plant licensing basis.

E. PAM Instrumentation

1. The primary PAM instrument functions sections evaluated were:

Remote Shutdown Indication (Technical Specification Table 4.3-10)
Containment Pressure (Table Item 1)
Reactor Coolant Hot Leg Temperature (Table Item 3)
Wide Range Pressurizer Pressure (Table Item 4)
Uncompensated Pressurizer Level (Table Item 5)
Steam Generator Pressure (Table Item 6)
Wide Range Steam Generator Level (Table Item 7)
AFW Flow (Table Item 8)
RCS Subcooled Margin Monitor (Table Item 9)
Acoustic Valve Monitor (Table Item 10)
Main Feedwater Flow (Table Item 12)
Containment Water Level (Wide Range) (Table Item 13)

For each of the PAM instruments addressed, the calculated drift was determined to be acceptable for control of plant parameters to effect a safe shutdown. There are no PAM instrumentation setpoints associated with any automatic functions. The PAM pressure and level transmitters evaluated were produced by Rosemount. Analysis for Rosemount transmitters was based on manufacturer testing data described in Generic Safety Analysis Issues section. The analysis of the PAM-related RTDs was based on industry testing data as described in Generic Safety Analysis Issues section. The Acoustic Valve Monitor components were determined to not be subject to drift, and the Main Feedwater flow indication is not relied on to mitigate the consequences of any event in the safety analyses. These instruments are discussed further below. The methodology used to analyze drift is discussed in the Generic Safety Analysis Issues section. An evaluation of maintenance history indicates that the indication loops have operated satisfactorily over their lifetime and no adverse trends in operability have been identified. Problems requiring maintenance have been primarily identified by channel checks and observations. Monthly channel check criteria and applicable surveillance calibration tolerances incorporating the drift analyses were established to continue to provide reliable detection of degraded channel operation. Monthly channel checks provide indication of degraded instrument operation. As-found calibration data is evaluated as part of the channel calibration surveillance program. Therefore, surveillance procedures provide assurance that failed or degraded loop components are detected so that appropriate action can be taken.

The projected 30-month uncertainties caused by drift were evaluated to be acceptable for control of the plant to effect a safe shutdown. Therefore, the effect of the surveillance interval extension on safety is small, and would not invalidate any assumption in the plant licensing basis.

2. Acoustic Valve Monitor

None of the Acoustic Valve Monitor components are subject to drift mechanisms. The accelerometers and charge converters do not have adjustment or calibration features. Once installed, functional verification is performed via the surveillance every refueling outage. An evaluation of maintenance history indicates that the passive sensors have operated satisfactorily over their lifetime and no adverse trends in operability have been identified. The components are replaced at the end of qualified life or when shown to be unresponsive to stimulation. The readout and alarm module is calibrated by the manufacturer at the factory and no further calibrations are required. Therefore, the effect of the surveillance interval extension on safety is small, and would not invalidate any assumption in the plant licensing basis.

3. Main Feedwater Flow

The Main Feedwater Flow recorder provides flow indication and a permanent record of Main Feedwater Flow. Main Feedwater is not assumed to operate in any safety analyses in the event of an accident. These transmitters were recently replaced with Rosemount Model 3051 transmitters. Analysis shows that extrapolated drift for main feedwater flow indication does not adversely affect main feedwater indicator accuracy. Monthly channel check criteria continue to provide reliable detection of degraded channel operation. Therefore, the effect of the surveillance interval extension on safety is small, and would not invalidate any assumption in the plant licensing basis.

F. Radiation Monitoring Instrumentation (Technical Specification Table 4.3-3)

1. Containment Purge and Exhaust Isolation Radiation Monitoring System (Table Item J.a)

The Containment Purge and Exhaust Isolation Radiation Monitoring System provides gross indication of intense gamma radiation fields resulting from abnormal shutdown conditions. The 18-month calibration surveillance is performed prior to entering Mode 6, the only Mode for which the instruments are required. The surveillance verifies proper operation of the detection and alarm systems. Performance of this surveillance prior to entering Mode 6 will continue. Therefore, the effect of the surveillance interval extension on safety is small, and would not invalidate any assumption in the plant licensing basis.

2. Containment Area High Radiation Monitoring System (Table Item J.b)

The Containment Area High Radiation Monitoring System provide gross indication of intense gamma radiation fields resulting from abnormal operating conditions. The 18-month surveillance consists of placing a gamma source next to the detector. An evaluation of 18-month surveillance test calibration and maintenance data from 1983 to 1992 found that the radiation monitors have always been within calibration tolerances and have not required adjustments, except for two post-modification calibrations. Instrument performance monitoring between channel calibrations includes channel functional tests, channel checks, and by use of a "live zero" feature provided by an internal radiation

source. This feature causes a constant instrument deflection to verify the circuitry from the monitor to the display is working. These checks and design features provide measures to detect system degradation without depending on the channel calibration. The detection systems are accurate to approximately 20% of the reading, because the function of the monitors is to provide gross indication. The alarm setpoint has been set considering the inaccuracy of the monitor. Small changes due to drift will not affect the ability of the system to perform its safety function. Therefore, the effect of the surveillance interval extension on safety is small and would not invalidate any assumption in the plant licensing basis.

G. Containment Sump Level Alarm System (Technical Specification 4.4.6.1.b.)

The Containment Sump High Level Alarm System is a non-safety-related system designed to provide one of the available indications of excessive RCS leakage. An evaluation of 18-month surveillance test calibration and maintenance data from 1983 to 1992 for the Containment Sump High Level Alarm System was performed. This evaluation indicates that the switches have operated satisfactorily over their lifetime and no adverse trends in operability have been identified nor are expected over the extended surveillance period. Also, the system has always been within calibration tolerances and not required calibration. Although a calibration is performed on the equipment, the level sensors are mechanical switches which are not subject to instrument drift. Therefore, the effect of the surveillance interval extension on safety is small, and would not invalidate any assumption in the plant licensing basis.

H. Conclusion

These surveillance interval extensions have a small effect on plant safety and would not invalidate any assumption in the plant licensing basis. This is based on the extension not requiring any setpoint changes and plant parameter indications still being acceptable for control of plant parameters to effect a safe shutdown.

DETERMINATION OF SIGNIFICANT HAZARDS

The proposed change has been evaluated against the standards in 10 CFR 50.92 and has been determined to not involve a significant hazards consideration, in that operation of the facility in accordance with the proposed amendments:

1. *Would not involve a significant increase in the probability or consequences of an accident previously evaluated.*

The proposed change would extend surveillance intervals for Reactor Protective System (RPS), Engineered Safety Features Actuation System (ESFAS), Power-Operated Relief Valve (PORV), Low Temperature Overpressure Protection (LTOP), Remote Shutdown, Post-Accident Monitoring (PAM), Radiation Monitoring, and Containment Sump Level Instruments.

The purpose of the RPS is to effect a rapid reactor shutdown if any one or a combination of conditions deviates from a pre-selected operating range. The system functions to protect the core and the Reactor Coolant System pressure boundary. The purpose of the ESFAS is to actuate equipment which protects the public and plant personnel from the accidental release of radioactive fission products if an accident occurs, including a loss-of-coolant incident, main steam line break, or loss of feedwater incident. The safety features function to localize, control, mitigate, and terminate such incidents in order to minimize radiation exposure to the general public. The Post-Accident Monitoring instruments provide the Control Room operators with primary information necessary to take manual actions, as necessary, in response to design basis events, and to verify proper system response to plant conditions and operator actions. The purpose of the Remote Shutdown System is to provide plant parameter indications to operators on a Remote Shutdown Panel to be used while placing and maintaining the plant in a safe shutdown condition in the event the Control Room is uninhabitable. The indications are used to verify proper system response to plant conditions and operator actions. The LTOP System protects against Reactor Coolant System overpressurization at low temperatures by a combination of administrative controls and hardware. The hardware includes two Power-Operated Relief Valves with variable pressurizer pressure setpoints when operating in the LTOP operating parameter region. The Containment Sump High Level Alarm System provides an alarm in the Control Room for a containment sump to provide one of the available indications of excessive RCS leakage during normal plant operation. The Containment Area High Range Radiation Monitoring System provides an indication of high radiation levels in containment. The Containment Purge System actuates equipment to prevent the release of radioactive material to the environment in the event of a reactor coolant leak, a shielding failure, or a fuel pin failure when the reactor vessel head is removed.

The instruments in each of the systems described above are designed to be used in response to an accident. Failure of any of these systems is not an initiator for any previously evaluated accident. Therefore, the proposed change would not involve an increase in the probability of an accident previously evaluated.

Many of the instruments addressed in this license amendment request will have or have recently had a new brand of sensor installed. The effect of the increased surveillance interval with the new sensors was analyzed. The new sensors do not effect the physical design description of the plant, any design or functional requirements, or surveillances. The proposed Technical Specification change extending the surveillance interval from 18 months to a refueling interval (nominally 24 months, not to exceed 30 months) does not physically change the plant, change any design or functional requirements, or effect the surveillances themselves. Analysis has shown that no trip setpoints need to be changed, and operator indications will continue to be accurate for control of plant parameters to effect a safe shutdown. The equipment will continue to perform as designed to mitigate the consequences of accidents. Therefore, the proposed change would not involve a significant increase in the consequences of an accident previously evaluated.

Therefore, the proposed change would not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. *Would not create the possibility of a new or different type of accident from any accident previously evaluated.*

The proposed change to increase the interval for RPS, ESFAS, PORV, LTOP, Remote Shutdown, PAM, Radiation Monitoring, and Containment Sump Level instrument surveillances from 18 months to a refueling interval (nominally 24 months, not to exceed 30 months) does not involve a significant change in the design or operation of the plant. No hardware is being added to the plant as part of the proposed change. Some detector upgrades in specific plant systems to enhance the performance of those systems have been or will be performed. However, those upgrades were evaluated and deemed acceptable under 10 CFR 50.59 and are not part of this request. The Reactor Protective System, Engineered Safety Features Actuation System, Power-Operated Relief Valve, Low Temperature Overpressure Protection, Containment Sump Level, and Radiation Monitoring actuation setpoints will not be changed. Analysis has shown that the remote shutdown and PAM indications will continue to be accurate. The proposed change will not introduce any new accident initiators. Therefore, this change does not create the possibility of a new or different type of accident from any accident previously evaluated.

3. *Does operation of the facility in accordance with the proposed amendment involve a significant reduction in a margin of safety?*

The impact of the surveillance interval extension request was evaluated for each Technical Specification-related safety function for each of the RPS, ESFAS, PORV, LTOP, Remote Shutdown, PAM, Radiation Monitoring, and Containment Sump Level instruments addressed by this submittal. In all cases, parameters specified in the related accident analysis were determined to be unaffected by the surveillance interval extension, and no accident analyses limits required changes. The Reactor Protective System, Engineered Safety Features Actuation System, Power-Operated Relief Valve, Low Temperature Overpressure Protection, Containment Sump Level, and Radiation Monitoring actuation setpoints will not be changed. Analysis has shown that the remote shutdown and PAM indications will continue to be accurate. The methods for detection of degraded instrument operation have not been changed, and remote shutdown and PAM operator indications will continue to provide adequate accuracy.

The proposed change does not affect the operation of the systems involved. The surveillance interval extension will not affect the design of the systems, and methods for detection of degraded instrument operation will continue to identify operation problems between calibrations. Therefore, the proposed change does not involve a significant reduction in a margin of safety.

ENVIRONMENTAL ASSESSMENT

The proposed amendment would change requirements with respect to the installation or use of a facility component located within the restricted area, as defined in 10 CFR Part 20, or changes to an inspection or surveillance requirement. We have determined that the proposed amendment involves no significant hazards consideration, and that operation with the proposed amendment would result in no significant change in the types or significant increases in the amounts of any effluents that may be released offsite, and in no significant increase in individual or cumulative occupational radiation exposure. Therefore, the

proposed amendment is eligible for categorical exclusion as set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment is needed in connection with the approval of the proposed amendment.

SCHEDULE

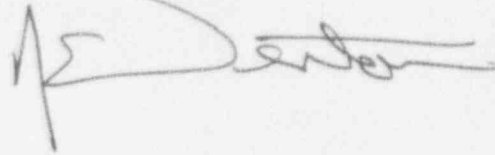
This change is requested to be approved and issued by March 31, 1996, to support incorporation of the changes during the 1996 Unit 1 refueling outage. Installation of the remaining Rosemount transmitters analyzed for this change is scheduled for the 1996 Unit 1 refueling outage.

SAFETY COMMITTEE REVIEW

These proposed changes to the Technical Specifications and our determination of significant hazards have been reviewed by our Plant Operations Safety Review Committee and Offsite Safety Review Committee, and they have concluded that implementation of these changes will not result in an undue risk to the health and safety of the public.

Should you have questions regarding this matter, we will be pleased to discuss them with you.


Very truly yours,



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

I hereby certify that on the 6th day of June, 1995, before me, the subscriber, a Notary Public of the State of Maryland in and for Calvert County, personally appeared Robert E. Denton, being duly sworn, and states that he is Vice President of the Baltimore Gas and Electric Company, a corporation of the State of Maryland; that he provides the foregoing response for the purposes therein set forth; that the statements made are true and correct to the best of his knowledge, information, and belief; and that he was authorized to provide the response on behalf of said Corporation.

WITNESS my Hand and Notarial Seal:


Notary Public

My Commission Expires:

2/2/98
Date

RED/JV/dlm

Attachments: (1) Unit 1 Technical Specification Marked-Up Pages
(2) Unit 2 Technical Specification Marked-Up Pages
(3) Index

cc: D. A. Brune, Esquire
J. E. Silberg, Esquire
L. B. Marsh, NRC
D. G. McDonald, Jr., NRC
T. T. Martin, NRC
P. R. Wilson, NRC
R. I. McLean, DNR
J. H. Walter, PSC

ATTACHMENT (1)

UNIT 1 TECHNICAL SPECIFICATION

MARKED-UP PAGES