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BGE

July 30, 1997

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
Request for Review and Approval of Reactor Pressure Vessels and Control Element Drive Mechanisms/Electrical System Report for License Renewal

**REFERENCES**:

(a) Letter from Mr. D. G. McDonald, Jr. (NRC) to Mr. R. E. Denton (BGE), dated January 2, 1996, "Updated Values for Pressurized Thermal Shock Reference Temperatures - Calvert Cliffs Nuclear Power Plant Unit Nos. 1 and 2," and attached NRC Safety Evaluation Report Pressurized Thermal Shock Evaluation

(b) Letter from Mr. R. E. Denton (BGE) to NRC Document Control Desk, dated August 18, 1995, Integrated Plant Assessment Methodology

(c) Letter from Mr. D. M. Crutchfield (NRC) to Mr. C. H. Cruse (BGE), dated, April 8, 1996, Final Safety Evaluation (FSE) Concerning The Baltimore Gas and Electric Company Report entitled, Integrated Plant Assessment Methodology

(d) Letter from Mr. S. C. Flanders (NRC), dated March 4, 1997, "Summary of Meeting with Baltimore Gas and Electric Company (BGE) on BGE License Renewal Activities"

This letter forwards the attached Integrated Plant Assessment (IPA) System Report on the Reactor Pressure Vessels and Control Element Drive Mechanisms/Electrical System for review and approval in accordance with 10 CFR Part 54, the license renewal rule. Should we apply for License Renewal, we will reference the IPA System Report on Reactor Pressure Vessels and Control Element Drive Mechanisms/Electrical System as meeting the requirements of 10 CFR 54.21(a), "Contents of

A001 1/1

application-technical information," and the demonstration required by 10 CFR 54.29(a)(1), "Standards for issuance of a renewed license."

Should we apply for License renewal, this report and Reference (a) satisfy 10 CFR 50.61(b)(1), which requires an updated assessment of the projected value of pressurized thermal shock "upon requests for a change in the expiration date for operation of the facility." In Reference (a), the NRC reviewed and accepted pressurized thermal shock projections for Calvert Cliffs for a period that includes the renewal period. Baltimore Gas and Electric Company requests that the NRC confirm that if an application is filed, this report, as part of that application, and Reference (a) satisfy 10 CFR 50.61(b)(1).

The information in this report is accurate as of the dates of the references listed therein. Per 10 CFR 54.21(b), an amendment or amendments will be submitted that identify any changes to the current licensing basis that materially affect the content of the license renewal application.

In Reference (b), Baltimore Gas and Electric Company submitted the IPA Methodology for review and approval. In Reference (c), the Nuclear Regulatory Commission (NRC) concluded that the IPA Methodology is acceptable for meeting 10 CFR 54.21(a)(2) of the license renewal rule, and if implemented, provides reasonable assurance that all structures and components subject to an aging management review pursuant to 10 CFR 54.21(a)(1) will be identified. Additionally, the NRC concluded that the methodology provides processes for demonstrating that the effects of aging will be adequately managed pursuant to 10 CFR 54.21(a)(3) that are conceptually sound and consistent with the intent of the license renewal rule.

In Reference (d), the NRC stated that if the format and content of these reports met the requirements of the template developed by BGE, the NRC could begin the technical review. This report has been produced and formatted in accordance with these guidance documents. We look forward to your comments on this and future reports as they are submitted and your continued cooperation with our license renewal efforts.

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, Charles H. Cruse, being duly sworn, state that I am Vice President, Nuclear Energy Division, Baltimore Gas and Electric Company (BGE), and that I am duly authorized to execute and file this response on behalf of BGE. To the best of my knowledge and belief, the statements contained in this document are true and correct. To the extent that these statements are not based on my personal knowledge, they are based upon information provided by other BGE employees and/or consultants. Such information has been reviewed in accordance with company practice and I believe it to be reliable.

Subscribed and sworn before me, a Notary Public in and for the State of Maryland and County of Calvert, this 30 day of July, 1997.

**WITNESS** my Hand and Notarial Seal: Michelle D. Hall
Notary Public

My Commission Expires: February 2, 1998
Date

CHC/SJR/dlm

Attachment (1): Appendix A - Technical Information 4.2 - Reactor Pressure Vessels and Control Element Drive Mechanisms / Electrical System

cc: R. S. Fleishman, Esquire
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ATTACHMENT (1)

# APPENDIX A - TECHNICAL INFORMATION

# 4.2 - REACTOR PRESSURE VESSELS

# AND CONTROL ELEMENT DRIVE MECHANISMS / ELECTRICAL SYSTEM


**Baltimore Gas and Electric Company**
**Calvert Cliffs Nuclear Power Plant**

**July 30, 1997**

## 4.2 Reactor Pressure Vessels and Control Element Drive Mechanisms/Electrical System

This is a section of the Baltimore Gas and Electric Company (BGE) License Renewal Application (LRA), addressing the Reactor Pressure Vessels (RPVs) and Control Element Drive Mechanisms (CEDMs)/Electrical System, including the Reactor Vessel Level Monitoring System (RVLMS). The RPVs and CEDMs/Electrical System was evaluated in accordance with the Calvert Cliffs Nuclear Power Plant (CCNPP) Integrated Plant Assessment (IPA) Methodology described in Section 2.0 of the BGE LRA. These sections are prepared independently and will, collectively, comprise the entire BGE LRA.

### 4.2.1 Scoping

System level scoping describes conceptual boundaries for plant systems and structures, develops screening tools that capture the 10 CFR 54.4(a) scoping criteria, and then applies the tools to identify systems and structures within the scope of license renewal. Component level scoping describes the components within the boundaries of those systems and structures that contribute to the intended functions. Scoping to determine components subject to an aging management review (AMR) begins with a listing of passive intended functions, and then dispositions the component types as either only associated with active functions, subject to replacement, or subject to an AMR either in this report or another report.

Section 4.2.1.1 presents the results of the system level scoping, 4.2.1.2 the results of the component level scoping, and 4.2.1.3 the results of scoping to determine components subject to an AMR.

Representative historical operating experience pertinent to aging is included in appropriate areas to provide insight supporting the aging management demonstrations. This operating experience was obtained through key-word searches of BGE's electronic database of information on the CCNPP dockets, and through documented discussions with currently assigned cognizant CCNPP personnel.

#### 4.2.1.1 System Level Scoping

This section begins with a description of the system, which includes the boundaries of the system as it was scoped. The CEDMs/Electrical System section also contains the RVLMS description. The intended functions of each system are listed and are used to define what portions of the system are within the scope for license renewal.

System Description/Conceptual Boundary

The CCNPP Unit 1 and Unit 2 RPVs are major parts of each Reactor Coolant System (RCS). Each RCS has one RPV, one pressurizer, two steam generators, two reactor coolant loops, and four reactor coolant pumps. [Reference 1, Section 1.1] The RPVs are comprised of a removable head with multiple penetrations, four primary coolant inlet nozzles, two primary coolant outlet nozzles, upper, intermediate and lower shell courses, bottom head and vessel supports. Each vessel is approximately 503-3/4-inches high, with an inside diameter of 172 inches, and is an all-welded, manganese molybdenum steel plate and forging construction. [Reference 2, Section 1.1.1] The RPV is supported vertically and horizontally by three pads welded to the underside of the RPV primary nozzles. Each RPV support consists of a support foot welded to the primary nozzle; a socket bolted to the support foot (with cap screws); and a sliding bearing whose spherical crown fits into the socket and whose flat side sliding surface rests on a base plate. [Reference 1, Section 4.1.3] Figure 4-2 in the CCNPP Updated Final Safety Analysis Report (UFSAR) has a drawing of the RPVs. [Reference 1, Section 4.1.3.1]

Each RPV contains the reactor vessel internals (RVI) and associated reactor core (See Section 4.3 of the BGE LRA for the RVI IPA results). The reactor is controlled by a combination of a chemical shim (dissolved boric acid), and control element assemblies (CEAs) which are made of a solid boron carbide neutron absorber. The CEAs (i.e., four tubes in a square matrix plus a central tube) are connected together at their tops by a yoke, which is, in turn, connected to the CEDM extension shaft (there are some CEDMs that have two yokes attached). The CEDMs are designed to permit rapid insertion of the CEAs in the reactor core by gravity. [Reference 1, Section 1.2.3, 1.2.7.2]

The CEDMs are magnetic jack-type drives capable of withdrawing, inserting, holding, or tripping a CEA from any point within their 137-inch stroke. Originally, 65 CEDMs were mounted on flanged nozzles on top of the reactor closure head. Eight of those CEDMs were connected to partial length CEAs, which have been subsequently removed. [Reference 1, Section 3.3.4.1] Two of the eight CEDMs have been modified to house RVLMS probes. [Reference 2, Section 1.1.1] The CEDM housings comprise the motor assembly, the motor housing assembly, the coil stack assembly, the upper pressure housing assembly, the shroud and conduit assembly, the reed switch assembly, and the drive shaft. [Reference 2, Section 1.1.3] The CEDM pressure housings are an extension of the reactor vessel, providing a part of the reactor coolant boundary, and are, therefore, designed to meet the requirements of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel (B&PV) Code, Section III, Nuclear Vessels. [Reference 1, Section 3.2.3.6] Refer to Figure 3.3-15 in the UFSAR for a drawing of the CEDMs.

The RVLMS housings consist of only a motor housing assembly, an upper pressure housing assembly (modified from the CEDM design), the shroud, the flange adapter assembly, and a Heated Junction Thermocouple (HJTC) probe assembly. [Reference 2, Section 1.1.3] This system is capable of providing the plant operator with the information needed to assess void formation in the reactor vessel head region, and the trend of liquid level in the reactor vessel plenum. The HJTC system is composed of two redundant channels, each powered from separate, reliable Class 1E sources. [Reference 1, Section 7.5.9.2]

System Operating Experience

The following are operating experiences related to the RPVs and CEDMs with the potential for affecting the intended functions of the components or systems.

Reactor Coolant Pump (RCP) Suction Deflector Failures

The 1988 and 1996 failures of the RCP suction deflector bolting at CCNPP are examples of changes to the RPV operating environment that could have had an effect on the ability of the RPV to perform its intended functions. A portion of the failed bolt was not recovered at the pump in each case, and was assumed to be lodged in the RPV on the cladding and near the downcomer. Combustion Engineering (CE) analyzed the 1988 failure and the potential impact to the Unit 2 RPV cladding of leaving the bolt until the 1989 refueling outage. CENC-1849, "Evaluation of Reactor Vessel Potential Wear of Bottom Head Clad Due to Loose Bolt," documents the analysis results. The evaluation determined there will be no unacceptable impact to the clad. The reactor core was off-loaded, the RVI removed, and bolt fragments were recovered during the 1989 refueling outage. Calvert Cliffs' Corrective Action Program results document the root cause analysis and the potential impact to the clad for the 1996 failure. The next scheduled RPV Inservice Inspection (ISI) will examine the affected area and recover the fasteners from the 1996 failure. [References 3 and 4]

RPV Closure Head Stud Overtensioning

On several occasions, closure studs were overtensioned during RPV head boltup, which could have had an effect on the RPV's intended function. Baltimore Gas and Electric Company's Design Engineering and CE evaluated each situation and determined that the studs and flanges were undamaged. For example, CR-9417-CSE92-1108, "Evaluation of a Reactor Vessel Closure Head Stud Elongation for BGE Calvert Cliffs," documents CE's evaluation that one of these incidents did not change the RPV's ability to perform the intended functions. Baltimore Gas and Electric Company's Design Engineering reviewed and accepted this evaluation. [Reference 5]

RCS Pressure Boundary Leakage

Calvert Cliffs experienced primary pressure boundary leakage of Alloy 600 Components caused by primary water stress corrosion cracking (PWSCC). Calvert Cliffs has a total of 244 Alloy 600 penetrations in the Unit 1 RCS, and 126 remaining in the Unit 2 RCS (120 pressurizer heater sleeves were replaced with Alloy 690 in 1989-1990). In 1989, a pressurizer vapor space instrument nozzle was found leaking, which led to the replacement of all four of the Unit 2 pressurizer vapor space nozzles. Also in 1989, 22 heater sleeves were found leaking, which resulted in the subsequent replacement of 119 heater sleeves with Alloy 690. During the 1994 Unit 1 refueling outage, two heater sleeves were found leaking, resulting in the nickel plating of the remaining 118 heater sleeves during the outage. The replacement and repair of the heater sleeves corrected the leakage problems and also contributed to the development of the CCNPP Alloy 600 Program Plan. Calvert Cliffs determined that the pressurizer heater sleeve leakage events were attributed to PWSCC, driven by residual stress created by cold working during fabrication. [Reference 6, Sections 1, 2, 3, 19] The Alloy 600 Program Plan is also important for RPVs and CEDMs/Electrical System components.

RCS Resin Intrusion

Calvert Cliffs Unit 1 had a resin intrusion in March 1989, and Unit 2 suffered a resin intrusion in January 1983, due to a failed outlet retention element of the ion exchanger in the purification system. The effect on the RPVs was evaluated at the time of the intrusions and found to be acceptable. Resin intrusions are a potential issue since resin decomposition products may contribute to cracking of sensitized Alloy 600. The Unit 1 resin intrusion event caused high sulfate levels in the RCS, which subsequently resulted in the Unit being shut down. The sulfate concentration in the RCS was evaluated by CE and BGE. The evaluation concluded that the potential increase for stress corrosion cracking (SCC) was insignificant. [Reference 7]

RPV Head Flange Corrosion

During the outage in 1990, corrosion (rust) was found on the Unit 2 RPV upper head, flange, and stud holes. The corrosion on the RPV upper head, flange, and stud holes was due to moisture collecting under a tent in the Containment Building. The corrosion was removed and the stud holes cleaned. An analysis determined the maximum allowed stud hole diameter. Measurements by BGE showed that the RPV flange was within its design tolerances, and that the RPV flange stud hole diameters were less than the maximum allowed diameter.

Unintentional Inclusion of Slag Stringer in RPV

The unintentional inclusion of a small slag stringer in a Unit 1 RPV weld is an example of a fabrication flaw with potential to impact the ability of the RPV to perform its intended functions. The slag stringer was identified during pre-operational ultrasonic technique examinations. Southwest Research Institute evaluated the stringer and found it to be within acceptable code limits. The Southwest Research Institute report, "Preliminary Ultrasonic Examination of Weld Seams in the Calvert Cliffs Unit 1 Nuclear Reactor Vessel," contains the examination and evaluation results. During the first Unit 1 RPV ISI, CCNPP performed enhanced ultrasonic technique examinations of the stringer to determine the acceptability of the flaw. The flaw was still within the acceptance limits of ASME Section XI. The 1986 Inservice Examination Report documents the results of this examination. [References 8 and 9]

RPV Surveillance Program

To address neutron induced embrittlement of the RPV materials, the CCNPP Comprehensive Reactor Vessel Surveillance Program (CRVSP) was formed. The CRVSP evaluates testing on RPV surveillance materials to address chemistry variability issues, and uses the surveillance and other research results to support industry development of measuring fracture toughness for surveillance materials. To increase the data on material properties for the CCNPP RPVs, BGE installed ex-vessel dosimetry, a supplemental CCNPP Unit 1 surveillance capsule, and purchased portions of the Shoreham RPV. The CRVSP also incorporates test results of the surveillance capsules from other power plants with similar material chemistry for analysis of CCNPP RPV fracture toughness. [Reference 10]

Low Temperature Overpressure Protection (LTOP)

The original CCNPP design did not require automatic protection from low temperature overpressurization events. Early industry overpressurization incidents resulted in regulatory changes that required greater protection, including automatic actions. To compensate for the limited pressure relieving capacity provided by the original design, BGE made commitments to the NRC, in July 1977, to enact certain administrative controls during low temperature operations. During 1989 to 1990, CCNPP re-evaluated all docketed commitments and incorporated them into site procedures. The LTOP Controls Report captures the LTOP design basis requirements and the administrative controls, including procedural controls. Calvert Cliffs' EN-1-214, "Low Temperature Overpressure Protection," is the controlling procedure for the preparation, review, and updating of the LTOP Controls Report. EN-1-214 also contains the requirements for periodic audits of LTOP controls. [References 11 and 12]

In summary, these events demonstrate that CCNPP has and will continue to address and perform corrective actions, as required, so that the RPVs and CEDMs/Electrical System components are capable of performing their intended function under the current licensing basis (CLB) conditions during the period of extended operation.

System Interfaces

The CEDM components under review in this report interface with the RCS. The RPV is a major component of the RCS. The RCS is within scope for license renewal and described in Section 4.1 of BGE's LRA. The RPVs and CEDMs/Electrical System also interfaces with the RVI (Section 4.3 of BGE's LRA), Cranes and Fuel Handling (Section 3.2 of BGE's LRA), and the containment interior structures (in Structures report, Section 3.3 of BGE's LRA). The leakage monitoring instrument line that is attached to the leakage monitoring tube from the RPV is another interface that is within scope for license renewal (in RCS report, Section 4.1 of BGE's LRA).

The interior surfaces of the RPVs and CEDMs are completely wetted by the RCS environment. Section 4.1 of BGE's LRA credits primary chemistry control as an aging management program to manage plausible aging of components in the RCS. Because this chemistry program is credited for the RCS, and because the interior surfaces of the RPVs and CEDMs are totally wetted from the RCS environment, the demonstration of the primary chemistry control program as an aging management program is not repeated in this section. Instead, the aging evaluation for the internal surfaces of the RPVs and CEDMs credits the chemically-treated and controlled, demineralized water environment provided by the RCS as an initial condition of this evaluation.

System Scoping Results

The RPVs, RVLMS, and system components are within scope for license renewal based on 10 CFR 54.4(a). In accordance with Section 4.1.1 of the CCNPP IPA Methodology, a detailed list of system intended functions was determined based on the requirements of 10 CFR 54.4(a)(1) and (2): [Reference 2, Section 1.1.4]

- To vent the RCS when natural circulation flow has been disrupted or blocked by accumulation of non-condensable gases;
- To provide reactor vessel coolant inventory level indication;
- To maintain the pressure boundary of the system (liquid and/or gas); and
- To provide structural support for the fuel assemblies, CEAs, and incore instrumentation (ICI) so that they maintain the configuration and flow distribution characteristics assumed in the CCNPP UFSAR Chapter 14 analyses.

The CEDMs and Electrical System components are within scope for license renewal based on 10 CFR 54.4(a). In accordance with Section 4.1.1 of the CCNPP IPA Methodology, a detailed list of system intended functions was determined based on the requirements of 10 CFR 54.4(a)(1) and (2): [Reference 13, Table 1]

- Provide a pressure retaining boundary for the RCS; and
- To provide rapid shutdown of the reactor.

The following CEDM intended functions were determined based on the requirements of 10 CFR 54.4(a)(3): [Reference 13, Table 1]

- For fire protection (§50.48) - Interrupt CEDM Motor Generator set output power to ensure safe shutdown in the event of a severe fire.
- For anticipated transient without scram (§50.62) - Initiate reactor trip by interrupting power to the CEDMs upon Diversified Scram System signal.
- For station blackout (§50.63) -Trip reactor to provide for rapid shutdown of the reactor.

The design parameters for each of the major RCS components are given in Section 4.1.3 of the CCNPP UFSAR. The RCS (and thus RPVs) is designated a Class I system for seismic design, and is designed for the criteria for load combinations and stress that are presented in Table 4-8 and design Codes, as listed in Table 4-9 of CCNPP UFSAR Section 4.1.3. The regulations listed in 10 CFR 54.4(a)(3) do not necessarily require nuclear safety grade components in order to respond to the requirements of the regulations. The components of the CEDMs that have intended functions listed above are subject to the applicable loading conditions identified in UFSAR Section 4.1.3, Table 4-8.

### 4.2.1.2 Component Level Scoping

Each RPV is identified by a single unique equipment identifier; therefore, the component level scoping for the RCS identified those two components as within scope for license renewal. These represent a single device type and equipment type. The RCS scoping also identified the RVLMS probes as within scope of license renewal; however, since these are directly related to the CEDM evaluation, they are addressed in this section of the BGE LRA. Each CEDM is also assigned a single unique equipment identifier; therefore, the component level scoping for the CEDMs/Electrical System identified 130 CEDMs. [Reference 2, Section 2.2] Based on the intended functions listed above, the portions of the RPVs and CEDMs/Electrical System that are within scope for license renewal include the following eight device types: [Reference 2, Table 2-1]

| Device Description | Device Code |
|---|---|
| 1. Control Element Drive Mechanism | (CEDM) |
| 2. Pressure Vessel | (PZV) |
| 3. 480 VAC Motor | (MB) |
| 4. 125/250 VDC Motor | (MD) |
| 5. Electrical Panel | (PNL) |
| 6. Test Point (RVLMS Probe) | (TP) |
| 7. Control Element Assembly | (CEA) |
| 8. Load Contactor | (CONT) |

### 4.2.1.3 Components Subject to Aging Management Review

This section describes the components of the RPVs and CEDMs/Electrical System which are subject to an AMR. It begins with a listing of passive intended functions and then dispositions the component types as either only associated with active functions, subject to replacement, evaluated in other reports, evaluated in commodity reports, or evaluated for aging management in this report.

Passive Intended Functions

In accordance with CCNPP IPA Methodology Section 5.1, the following RPV functions were determined to be passive. [Reference 2, Table 3-1a]

- To maintain the pressure boundary of the system (liquid and/or gas); and
- To provide structural support for the fuel assemblies, CEAs, and ICI so that they maintain the configuration and flow distribution characteristics assumed in the UFSAR Chapter 14 analyses.

Similarly, the following CEDMs/Electrical System function was determined to be passive. [Reference 2, Table 3-1b]

- Provide pressure boundary for the reactor coolant system.

The components of the RPVs and CEDMs/Electrical System, and their supports, were reviewed, and all of the components that have at least one intended function were identified. Of the eight device types identified within scope for license renewal: [Reference 2, Table 3-2]

- 480 VAC Motors, 125/250 VDC Motors, Control Element Assemblies, and Load Contactors are only associated with active functions.
- Electrical panels in the CEDMs/Electrical System are evaluated for the effects of aging in the Electrical Panels Commodity Evaluation in Section 6.2 of BGE's LRA.
- Electrical components and cables associated with components in the system are evaluated for the effects of aging in the Environmental Qualification Commodity Evaluation in Section 6.3 of BGE's LRA.

The three remaining device types, listed in Table 4.2-1, are subject to an AMR, and are the subject of the remainder of this report.

**TABLE 4.2-1**

**RPVs AND CEDMs / ELECTRICAL SYSTEM**
**DEVICE TYPES REQUIRING AMR**

| |
|---|
| RPV (PZV) |
| CEDM |
| RVLMS Test Point (TP) |

Other RPV Subcomponent Intended Functions

The RCS System Level Scoping identified the intended functions of the RPVs, and the CEDMs/Electrical System Component Level scoping identified those of the CEDMs/Electrical System. These results were presented in Section 4.2.1.1 under System Scoping Results. Using the CCNPP IPA Methodology of Section 6.2.2, Structures and Components Grouping, the RPVs were further subdivided into individual subcomponents. These groups of RPV subcomponents were found to have additional passive intended functions beyond that of RCS pressure-retaining boundary. These additional passive intended RPV functions are addressed throughout this section. Some of the following RPV passive functions contribute to the RVIs' performance of its intended functions. The additional passive intended functions determined in accordance with CCNPP IPA Methodology Section 6.2.2 are listed below: [Reference 2, RPV, Attachment 4]

- Support RPV, restrict lateral vessel movement, allow vessel thermal expansion and contraction [Vessel Supports (i.e., sliding plate, plate steel, anchor bolts, cooling jacket, sockets, cap screws, shim and base plates, and support pads)];
- Monitor leakage from between the O-rings [Leakage Monitoring Tube];
- Minimize thermal stress to the CEDM [CEDM Thermal Sleeves];
- Reduce core inlet flow inequalities and prevent formation of large vortices [Flow Skirt];
- Prevent excessive core displacement under specified accident conditions [Core Stop Lugs];
- Limit flow-induced vibrations in the core support barrel [Core Stabilizing Lugs, Bolts, and Shims, Snubbers Spacer Blocks and Capscrews]; and
- Support surveillance capsules, which provide supporting information to predict the embrittlement condition of the RPV [Surveillance Capsule Holders].

Baltimore Gas and Electric Company may elect to replace components for which the AMR identifies further analysis or examination is needed. In accordance with the License Renewal Rule, components subject to replacement based on qualified life or specified time period would not be subject to an AMR.

### 4.2.2 Aging Management

The potential ARDMs for the RPVs and CEDMs/Electrical System components are listed in Table 4.2-2. The plausible ARDMs are identified in the table by a check mark (✓) in the appropriate column. Those potential ARDMs, which were considered in the analysis to not be plausible, are marked with an (x) in the Not Plausible For System Column. [Reference 2, Tables 4-1 and 4-4] The device types listed in Table 4.2-2 are those previously identified in Table 4.2-1 as passive and long-lived. The device types not included in Table 4.2-2 were previously dispositioned with the CCNPP IPA Methodology as performing an active function and/or addressed in commodity evaluations. For efficiency in presenting the results of these evaluations in this report, the components here are grouped together based on similar ARDMs. [Reference 2, Section 4.4]

## TABLE 4.2-2

## POTENTIAL AND PLAUSIBLE ARDMs FOR THE RPVs AND CEDMs / ELECTRICAL SYSTEM

| Potential ARDMs | Device Types | | | Not Plausible for System |
|---|---|---|---|---|
| | PZV (RPV) | TP (RVLMS) | CEDM | |
| Corrosion Fatigue | | | | x |
| Erosion | | | | x |
| Fatigue (Group 3) | ✓(a) | ✓ | ✓ | |
| General Corrosion (Group 1) | ✓(b) | | | |
| Hydrogen Embrittlement | | | | x |
| Neutron Embrittlement (Group 4) | ✓(c) | | | |
| Stress Corrosion Cracking (Group 5) | ✓(d) | | | |
| Stress Relaxation | | | | x |
| Wear (Group 2) | ✓(e) | ✓(e) | ✓(e) | |

✓ Indicates plausible ARDM determination.

( ) Indicates that not all components of a device type are affected by the ARDM. The notes below clarify the exceptions. [Reference 2, Attachments 4, 5, and 6]

(a) All RPV components except for vessel supports (sliding bearing, plate steel, anchor bolts, cooling jacket, sockets, capscrews, shim and base plates, and support pads) and snubber spacer blocks and capscrews.

(b) Only the unclad external surfaces of the RPV upper/lower head and cylindrical shell plates and their welds, nozzle welds; RPV and closure head flanges, inlet and outlet nozzles, and nozzle safe ends; RPV closure head studs, nuts, and washers; RPV vessel supports (plate steel, anchor bolts, cooling jacket, sockets, shim and base plates, and support pads).

(c) Only the vessel plates and welds of the lower shell, intermediate shell, and lower portion of the nozzle shell courses.

(d) Only the RPV leakage monitoring tube; ICI tube nozzles, Vent pipe, and CEDM nozzles; flow skirt; core stop lugs; core stabilizing lugs; surveillance capsule holders; and RPV supports anchor bolts.

(e) Only the vessel flanges (threaded stud holes); RPV closure head studs, nuts, and washers; core stabilizing lugs and snubber spacer blocks; RPV ICI tube flanges including their bolts and nuts; and RVLMS blind flange adapter hub vent plug and flange nut, Grayloc clamp set, studs, nuts and seal plug drive nut; CEDM ball seal housing; and CEDM and RVLMS upper housing assembly steel balls.

## APPENDIX A - TECHNICAL INFORMATION
## 4.2 - REACTOR PRESSURE VESSELS AND CONTROL ELEMENT DRIVE MECHANISMS / ELECTRICAL SYSTEM

The following discussions present information on plausible ARDMs. The discussions are grouped by ARDM and address the materials and environment pertinent to the ARDM, the aging effects for each plausible ARDM, the device types that are affected by each, the methods to manage aging, the aging management program(s), and the aging management demonstration. The groups addressed are:

Group 1 - general corrosion;

Group 2 - wear;

Group 3 - fatigue;

Group 4 - neutron embrittlement; and

Group 5 - stress corrosion cracking.

### Group 1 (general corrosion) - Materials and Environment

Table 4.2-2 shows that corrosion is plausible for only specific RPV components. This ARDM affects the unclad external surfaces of carbon steel RPV components that could be wetted by boric acid or component cooling water. The group of subcomponents potentially affected and material characteristics are: [Reference 2, RPV Attachments 4, 5, and 6]

- RPV upper/lower head/cylindrical shell plates (SA-533 Grade B Class 1 with stainless steel cladding on internal surfaces) and their welds and stainless steel clad;
- RPV and closure head flanges, inlet and outlet nozzles, nozzle safe ends (SA-508-64 Class II with stainless steel cladding on internal surfaces);
- RPV closure head studs, nuts, and washers (A-540 Grade 23 Class III and A-540 Grade 24 Class III),
- RPV supports: plate steel (A-302 Grade B), anchor bolts (A-354 Grade B6), cooling jacket (A-106 Grade B), sockets (A-536), shims and base plates (ANSI 4140), and support pads (SA-508-64); and
- RPV nozzle welds with stainless steel cladding on internal surfaces.

The internal environment of the RPVs and CEDMs is that of the RCS, which contains water at an operating pressure of approximately 2250 psia. Normal RCS operating temperatures are approximately 548°F in the cold leg and 599.4°F in the hot leg. The RCS maintains a flow rate of approximately $134 \times 10^6$ lbm/hr. [Reference 1, Section 4.1.1, Table 4-1]. The external environment is ambient atmospheric air inside the Containment Building that is climate controlled. This environment in the Containment Building during normal operations has maximum humidity of 70% and maximum temperature of 120°F. [Reference 1, Table 9-18] The RCS also contains chemicals for controlling reactor power (boric acid) and corrosion control.

### Group 1 (general corrosion) - Aging Mechanism Effects

Corrosion is degradation that results in wall thinning due to oxidation of low alloy and high alloy ferritic steel components. Forms of general corrosion include uniform attack, pitting, and intergranular attack. [Reference 2, Attachment 7]

General corrosion is a concern for RPV low alloy and high alloy ferritic steel subcomponents listed above. Therefore, corrosion is a plausible ARDM for the RPV steel subcomponents. Corrosion is concern due to the boric acid concentration of the RCS fluid. The particular concern is potential leakage onto external component surfaces from this RCS fluid. There has also been corrosion observed in the O-ring grooves of the RPV head due to a high local dissolved oxygen content. [Reference 2, RPV, Attachment 6, Code T]

If left unmitigated in the long-term, corrosion could eventually result in sufficient wall thinning from localized pitting, and/or general area material loss to cause failure of the pressure-retaining capability under CLB design loading conditions.

### Group 1 (general corrosion) - Methods to Manage Aging

Mitigation: The effects of corrosion cannot be completely prevented, but they can be mitigated by minimizing the exposure of the carbon steel surfaces of the RPV metal components to an aggressive chemical environment. Stainless steel cladding on the interior of the RPV components minimizes the effects of general corrosion on the interior surfaces exposed to reactor coolant. However, mitigation of corrosion on the exterior surfaces of the RPV requires the prevention of RCS leakage from the RPV pressure boundary, and the removal of any boric acid residue from exterior RPV surfaces.

Discovery: The effects of corrosion on the RPV pressure boundary can be discovered through a program of visual inspections on the RPV areas susceptible to this ARDM. Inspection of the areas around the RPVs could identify leakage occurring, and result in corrective actions being taken before corrosion could degrade the RPV's intended function. The inspections must be performed on a frequency that is sufficient to ensure that the minimum vessel thickness requirements will be met until the next inspection is performed.

### Group 1 (general corrosion) - Aging Management Program(s)

Mitigation: The CCNPP "Boric Acid Corrosion Inspection Program," (MN-3-301) can mitigate the effects of boric acid corrosion through discovery of leakage of RPVs and CEDMs/Electrical System components, and removal of any boric acid residue that is found. Removal of any boric acid leakage from component surfaces mitigates the effects of this substance on these surfaces. [Reference 2, Attachment 8, General Corrosion]

Discovery: The CCNPP ISI Program is credited with discovering general corrosion on the RPV supports and anchor bolts. These RPV supports and anchor bolts will be visually examined, as defined in IWF-2500 of the ASME Code Section XI and acceptance criteria contained in IWF-3410. The purpose of the ISI Program is to control the methods and actions for ensuring the structural and pressure-retaining integrity of safety-related nuclear power plant components, in accordance with the rules of ASME Section XI. [Reference 14, Section 3.0D] The Long Term Plan uses the requirements of Section XI of the ASME Code, 1983 Edition through Summer 1983 Addenda, and is subject to periodic update per 10 CFR 50.55a. [Reference 15, Section 1.2.1]

The scope of the existing ISI Program for the RPVs and CEDMs includes examination and inspection of components identified in ASME Section XI, Subsection IWB. [Reference 16, Section 1.2A] The ISI

Program is performed to meet the requirements of references identified in Section 1.2A of Reference 16. An extensive list of the developmental and performance references for the existing ISI Program is provided in Section 2.0 of Reference 16.

Inservice inspection requirements in ASME Section XI, as implemented by the existing ISI Program, provide for visual examination of accessible surfaces of reactor vessel components. [Reference 17, Table IWB-2500-1] For visual examination of the RPV supports and anchor bolts, ASME Section XI requires determining the general mechanical and structural conditions of the components for wear, and loss of integrity at bolted connections. Examinations may require, as applicable, determination of structural integrity, measurement of clearances, detection of physical displacements, structural adequacy of supporting elements, connections between load-carrying structural members, and tightness of bolting. [Reference 17, IWA-2213 Visual Examination VT-3]

If any abnormal condition is identified, the ASME Code provides requirements for the timely correction of the condition. [Reference 17, IWA-4130 Repair Program] Visual inspections can readily identify damage to the RPV supports and anchor bolts, such as would be caused by general corrosion, wear, and other aging mechanisms. The corrective actions taken will ensure that the RPV supports and anchor bolts remain capable of performing their intended functions under all CLB conditions.

The ISI Program is subject to internal and independent assessments, and is recognized through these assessments as performing highly effective examinations and aggressively pursuing continuous improvements. Baltimore Gas and Electric Company monitors industry initiatives and trends in the area of ISI and non-destructive examination, and plays a leadership role in developing, analyzing, and advancing non-destructive examination and ISI methods. The program is also subject to frequent external assessments by the Institute for Nuclear Power Operations, NRC, and others.

Operating experience relative to the ISI Program at CCNPP has been such that no site-specific problems or events have required changes or adjustments. The program has been effective in its function of performing examinations required by ASME Section XI.

Two design improvements have been made to the reactor vessel to facilitate the ISI Program. First, pads have been placed on the outside of the vessel to function as location benchmarks for ultrasonic inspection. Second, additional room has been provided between the nozzle piping and surrounding concrete to allow inspection of the piping. [Reference 1, Section 4.1.5.5]

The RPV support components should not be subject to an aggressive environment. However, the supports are normally not accessible, and there is the potential for boric acid leakage and component cooling water (which cools the supports) leakage creating an adverse environment. Therefore, the supports are examined on a regular basis (per ASME Section XI) to ensure they are not subject to such an environment. [Reference 2, Attachment 6, Code T]

Discovery of boric acid leakage is performed by MN-3-301, Maintenance Procedures RV-22, "RPV O-Ring Replacement," and RV-62, "RPV Stud, Nut, and Washer Cleaning and Inspection." These programs and procedures require the visual inspection of RPVs and CEDMs/Electrical System components for boric acid leakage. MN-3-301 requires investigation of any leakage that is found, and implements the visual examination of external surfaces with the system under normal operating pressure (i.e., VT-2) for the RPVs and CEDMs/Electrical System components in accordance with ASME Section XI IWA-2212. The Maintenance Procedures supplement the Boric Acid Corrosion Inspection Program through inspections of their respective RPV components. [Reference 2, Attachment 8, General Corrosion]

The basis for the establishment of the program is GL 88-05, "Boric Acid Corrosion of Carbon Steel Reactor Pressure Boundary Components in PWR Plants." The ISI Program controls examination and test methods and actions to minimize the loss of structural and pressure-retaining integrity of RCS pressure boundary components due to boric acid corrosion. [Reference 16, Section 3.0.C; Reference 18, Section 1.1]

The scope of the program is threefold: (1) It provides examination locations where leakage may cause degradation of the primary pressure boundary by boric acid corrosion; (2) provides examination requirements and methods for the detection of leaks; and (3) provides the responsibilities for initiating engineering evaluations and the subsequent proposed corrective actions. [Reference 18, Section 1.2]

Upon reaching reactor shutdown, ISI personnel are required to perform a containment walkdown inspection (VT-2) as soon as possible after attaining hot standby condition to identify and quantify any leakage found in specific areas of the Containment Building. A second ISI walkdown is performed prior to plant startup. The ISI must ensure that all components that are the subject of Issue Reports (IRs), where boric acid leakage has been found, are examined in accordance with the requirements of this program. [Reference 18, Sections 5.1 and 5.2] Calvert Cliffs procedure QL-2-100, "Issue Reporting and Assessment," defines requirements for initiating, reviewing, and processing IRs, and resolution of issues. Issue reports are generated to document and resolve process and equipment deficiencies and nonconformances. [Reference 19, Sections 1.1 and 1.2]

Under the Boric Acid Corrosion Inspection Program, the VT-2 walkdown examinations must be performed in accordance with ASME XI, IWA-2212. The VT-2 walkdown examinations must include the accessible, external, exposed surfaces of pressure-retaining, noninsulated components; floor areas or equipment surfaces located underneath noninsulated components; vertical surfaces of insulation at the lowest elevation where leakage may be detected, and horizontal surfaces at each insulation joint for insulated components; floor areas and equipment surfaces beneath components and other areas where water may be channeled for insulated components whose external insulation surfaces are inaccessible for direct examination; and for discoloration or residue on any surface for evidence of boric acid accumulation. Any leakage detected must be reported on an IR for corrosion degradation assessment. [Reference 18, Section 5.2]

Issue reports that have been written in accordance with this program are required to address: (1) The removal of the boric acid residue; and (2) The inspection of the affected components for general corrosion. If general corrosion is found on a component, the IR resolution provides an evaluation of the component for continued service and corrective actions to prevent recurrence. [Reference 18, Section 5.3]

In addition to the Boric Acid Corrosion Inspection Program, credit is also taken for CCNPP Procedure RV-62 for the discovery of general corrosion. RV-62 specifies the procedural steps and materials to be used in the cleaning and inspection of the RPV studs, nuts, and washers. The procedure describes the inspecting and reporting of studs, nuts, and washers for any damage that is found. [Reference 20, Section 6.2].

Maintenance Procedure RV-22 is also credited for the discovery of general corrosion on the RPV head and vessel O-ring flange sealing area. This procedure provides for inspection and acceptance criteria for minor pitting, nicks, and scratches near or on the O-ring sealing area. [Reference 21, Section 6.3] Any evidence of general corrosion would be found during the performance of this procedure.

The Boric Acid Inspection Program has evolved to account for operational experience. Reactor Coolant System leakage has been discovered that has affected the focus of subsequent inspections. For example: (1) A Unit 2 pressurizer heater sleeve was discovered to have a leak, and as a result, CCNPP instituted a more comprehensive inspection of the pressurizer heater sleeves in Unit 1; and (2) A seal vent line in containment developed a leak which dripped onto and caused surface corrosion of a RCS elbow resulting in increased attention with the program to such leaks.

Both CCNPP Units have had occurrences of boric acid leakage through the ICI flange connections. In March, 1993 (Unit 2) and February 1994 (Unit 1), evidence of boric acid leakage and corrosion were discovered on the ICI flanges and flange nuts. The cause was a change of gasket material which changed the required gasket crush force. Boric acid deposits were discovered on the ICI flanges and RPV head insulation, and the potential for wastage of the head material was noted. [Reference 22, page 2]

Additionally, the program has evolved with regard to the qualification level of personnel for evaluating boric acid leaks. The program dictates a minimum qualification level of non-destructive examination Level II Inspector for the evaluation of boric acid leaks. Any person conducting a walkdown or inspection may discover boric acid leakage. Such leakage would then be documented in an IR by the individual discovering the leak, and routed to the ISI group for closer inspection and evaluation by a Level II Inspector. This approach provides for wide boric acid leakage inspection coverage, but ensures boric acid leakage and its effects are evaluated by qualified individuals

The corrective actions taken as a result of IRs under this program and previously discussed programs will ensure that the RPV components remain capable of performing their intended function under all CLB conditions during the period of extended operation.

**Group 1 (general corrosion) - Demonstration of Aging Management**

Based on the material presented above, the following conclusions can be reached with respect to the corrosion of the RPV components:

- The RPV components provide the RCS pressure-retaining boundary and provide structural support to the RPV, while allowing limited motion for thermal expansion.
- General corrosion is plausible for RPV components listed here and could lead to loss of the pressure-retaining boundary or structural support function.
- The CCNPP Boric Acid Corrosion Inspection Program provides for examination of boric acid on carbon steel surfaces, and provides appropriate corrective action to prevent corrosion if boric acid leakage is found.

- The CCNPP ISI Program provides for the inspection of RPV support components per the requirements of ASME Section XI. Though general corrosion is not expected, the status of the components can be evaluated on a regular basis to ensure that these components are not subjected to an environment conducive to general corrosion.
- The CCNPP Procedure RV-62 provides for supplemental visual examination of the RPV studs, nuts, and washers.
- The CCNPP Procedure RV-22 provides for supplemental visual examination of the RPV head O-ring grooves and flange areas.
- Examinations will be performed, and appropriate corrective actions will be taken, if corrosion is discovered.

Therefore, there is reasonable assurance that the effects of corrosion on RPV components will be managed in order to maintain the components pressure boundary integrity under all design conditions required by the CLB during the period of extended operation.

**Group 2 (wear) - Materials and Environment**

Table 4.2-2 shows that wear is plausible for the RPV, CEDM, and RVLMS components. This group of components and their material composition are listed below: [Reference 2, Attachments 4, 5, and 6]

- RPV vessel flanges (threaded stud holes) (SA-508-64 Class II Forging);
- RPV head closure studs, nuts and washers (A-540 Grade 23 Class III and A-540 Grade 24 Class III);
- ICI tube nozzle flanges (SA-182, Type 316);
- ICI tube nozzle flange bolts and nuts (SB-637 or SA-453 Grade 660);
- RPV core stabilizing lugs (SB-166 Alloys 600 and X-750) and snubber spacer blocks (A-240-63). The snubber spacer blocks are hard faced with Stellite to minimize wear. [Reference 1, Figure 3.3.-12];
- RVLMS blind flange adapter hub vent plug and flange nut (SA-479 Type 316 stainless steel);
- CEDM ball seal housing (SA-479 Type 316 stainless steel);
- CEDM and RVLMS upper housing assembly steel balls (Type 440 stainless steel);
- Grayloc clamp set (SA-182 Type 304 stainless steel);
- Grayloc clamp studs and nuts (SA-453 Grade 660 Grade A or B with Chrome plating on stud threads); and
- Seal plug drive nut (SA-479 Type 347 stainless steel).

The internal environment of the RPVs and CEDMs is that of the RCS, which contains water at an operating pressure of approximately 2250 psia. The RCS maintains a flow rate of approximately $134 \times 10^6$ lbm/hr. [Reference 1, Section 4.1.1, Table 4-1] Certain RPV components are removed/disassembled and reinstalled for refueling outages. Many of these RPV components are tightly joined together to form the RCS pressure boundary.

**Group 2 (wear) - Aging Mechanism Effects**

Components such as the RPV closure head studs, nuts, and washers; the ICI tube nozzle flanges, studs, and nuts, and some of the internal components (core stabilizing lugs and snubber spacer blocks), are susceptible to mechanical wear due to the relative motion between them, and therefore, wear is a plausible ARDM. Mechanical wear is the deterioration of a surface due to material removal caused by the motion between contacting surfaces. [Reference 2, Attachments 5, 6, and 7] The Stellite facing on the snubber spacer block is a design feature that minimizes the effects of wear of these components. [Reference 1, Figure 3.3-12]

Long-term exposure to wear could lead to material loss and, if unmitigated, could eventually result in loss of the pressure-retaining capability and reduced ability to limit flow-induced vibrations in the core support barrel (core stabilizing lugs and snubber spacer block) under CLB design loading conditions. Therefore, mechanical wear was determined to be a plausible ARDM for which aging effects must be managed for areas of the RPVs and CEDMs/Electrical System components mentioned here.

**Group 2 (wear) - Methods to Manage Aging**

Mitigation: Mechanical wear on those components that are manipulated during refueling operations can occur; but, they usually are not subject to mechanical wear during normal operation. Minimizing the amount of component manipulation can mitigate wear. Those components that are normally not manipulated mitigate wear by proper design and material selection.

Discovery: With proper design, mechanical wear occurs slowly and over long periods of time, and is revealed as material loss of the components themselves. This wear can be discovered and monitored by visual inspection of the affected areas. Visual inspections of components can find any potential mechanical wear on the components. Wear of CEDM and RVLMS vent balls will lead to minor leakage and an indication of boric acid leakage at the normally inaccessible CEDM and RVLMS vent balls. This will lead to a more thorough examination of the components and, if required, will result in their being replaced. [Reference 2, CEDMs, Attachment 2].

Indications of wear identified during visual examinations of RPVs and CEDMs/Electrical System components during refueling outages must be recorded and evaluated for potential damage. Evidence of mechanical wear that will compromise the components intended functions before the next inspection must be repaired.

**Group 2 (wear) - Aging Management Program(s)**

Mitigation: There are no programs credited for the mitigation of wear beyond that provided by the original selection of materials and design of joints and interfaces, and by standard industry methods for tensioning and detensioning components.

## APPENDIX A - TECHNICAL INFORMATION
## 4.2 - REACTOR PRESSURE VESSELS AND CONTROL ELEMENT DRIVE MECHANISMS / ELECTRICAL SYSTEM

Discovery: Inservice inspection is the existing program designed to detect and manage the effects of wear for the RPVs and CEDMs/Electrical System components. Visual examination performed by the ISI Program will readily detect wear in the RPV studs, nuts, and washers; the RPV flanges (threaded stud holes); the core stabilizing lugs; and snubber spacer blocks. The mechanical maintenance procedures perform supplementary visual inspections for the ICI flanges and for the RPV studs, nuts, and washers. [Reference 2, RPV, Attachment 8] The Long Term ISI Plan implements the requirements of Section XI of the ASME Code, 1983 Edition through Summer 1983 Addenda. [Reference 15, Section 1.2.1] For a detailed description of the ISI process, refer to the discussion under General Corrosion-Aging Management Programs.

In addition to the ISI Program, credit is taken for CCNPP Procedure RV-62 for the discovery of wear. [Reference 2, RPV Attachment 8] This procedure specifies the procedural steps, materials, and acceptance criteria to be used in the cleaning and inspection of the RPV studs, nuts, and washers. The procedure describes what the inspection process should be looking for, and how to report any wear or damage that is found. RV-62 also lists the acceptance criteria for contact between load bearing surfaces as a minimum of 70 percent. [Reference 20, Section 6.2] Specific instructions are provided in the procedure for the operation of equipment used in the cleaning of RPV studs and nuts.

Calvert Cliffs procedure RV-85, "ICI Flange Cleaning and Inspection," is credited for the discovery of wear. [Reference 2, RPV, Attachment 8] The procedure refers to the inspection of the ICI flanges for scratches, nicks, steam cuts, gouges, or rolled metal, and the documentation of any findings of wear. [Reference 23, Section 6.2 and Attachment 1]

Calvert Cliffs procedure RVLMS-2, "Installation of the Flexible HJTC in the Reactor," is credited for the discovery of wear. RVLMS-2 will be modified to include statements that visual inspection of Grayloc clamps, the RVLMS flanges, the associated studs and nuts, and seal plug, and drive nut are to be performed each time the RVLMS housings are reassembled. Components of the RVLMS will be replaced as necessary, based on the results of the inspection. [Reference 2, CEDM, Attachment 8 and 10]

The Boric Acid Corrosion Inspection Program is also credited here for finding wear on the CEDM and RVLMS vent balls. During the boric acid inspection process, evidence of boric acid at these locations would indicate that the CEDM and RVLMS vent balls were experiencing wear and would need to be replaced. [Reference 2, CEDM, Attachment 8] For a description of the CCNPP Boric Acid Corrosion Inspection Program, refer to the discussion under Group 1 (general corrosion), Aging Management Programs.

The existing CCNPP Maintenance Procedures listed here have been in use to clean and inspect RPV components, and thus supplement the ISI Program by providing another means to find any degradation of these components before their intended function is compromised.

Inspections of the RPV studs, nuts, washers, and stud holes during every refueling outage ensures the closure components meet the design requirements. Stud removal from the RPV flanges has resulted in galling and thread damage on at least one occasion. Baltimore Gas and Electric Company analyzed the damaged threads and developed a repair modification package (FCR 82-0070) to install sleeves to restore the threaded connection integrity. The RPV design bases were reviewed and modified as part of this repair.

**Group 2 (wear) - Demonstration of Aging Management**

Based on the factors presented above, the following conclusions can be reached with respect to the wear of the RPVs and CEDMs/Electrical System components:

- The RPVs and CEDMs/Electrical System components provide a RCS pressure-retaining boundary and limit the flow induced vibrations in the core support barrel, so their integrity must be maintained under CLB design conditions.
- Mechanical wear is plausible for the components listed above, which could lead to material loss and reduced capability of the components to perform their passive intended function of maintaining the RCS pressure boundary and limiting vibrations.
- The CCNPP ISI Program provides for the inspection of the components listed above per the requirements of ASME Section XI. Though mechanical wear cannot be completely prevented, the status of the components can be evaluated on a regular basis and corrective actions can be taken as conditions indicate component wear.
- The CCNPP Boric Acid Corrosion Inspection Program provides for inspection around the CEDM and RVLMS vent areas. Inspection of these areas could indicate the presence of RCS leakage (dried boric acid) and the necessary replacement of vent balls due to wear.
- The CCNPP Procedure RV-62 involves the cleaning and inspection of the RPV studs, nuts, and washers, and thus performs supplementary visual inspection for the ISI Program.
- The CCNPP Procedure RV-85 involves the cleaning and inspection of the ICI tube nozzle flanges, and thus performs supplementary visual inspection for the ISI Program.
- The CCNPP procedure for installation of the HJTC, RVLMS-2, will be modified to include explicit visual inspection of the Grayloc clamps, RVLMS flanges, and the associated studs, nuts, and seal plug and drive nut. This modification will supplement the ISI Program for the surveillance of wear.

Therefore, there is reasonable assurance that the effects of wear will be managed in order to maintain the RPVs and CEDMs/Electrical System components' pressure boundary integrity and limiting flow induced vibrations in the core support barrel under all design conditions required by the CLB during the period of extended operation.

**Group 3 (low-cycle fatigue) - Materials and Environment**

Table 4.2-2 shows that fatigue is plausible for all RPV, CEDM, and RVLMS components under review in this section, except for RPV support components and snubber spacer blocks. The RPV was evaluated and low-cycle fatigue was determined to be plausible because of cyclic stress loads that result from normal operation and maintenance. The locations of interest for low-cycle fatigue are the RPV main coolant outlet nozzles and closure head flange studs. All other RPV components/subcomponents are considered to have low susceptibility to low-cycle fatigue. [Reference 2, RPV, Attachments 4, 5, and 6]

## APPENDIX A - TECHNICAL INFORMATION
## 4.2 - REACTOR PRESSURE VESSELS AND CONTROL ELEMENT DRIVE MECHANISMS / ELECTRICAL SYSTEM

The RPVs and CEDMs were designed in accordance with the ASME Section III, Nuclear Vessels (Class A) Winter 1967 Addenda to the ASME B&PV Code. [Reference 1, Section 4.1.1.1 and Tables 4-8 and 4-9] The RPV outlet coolant nozzles are fabricated from SA-508-64, Class II steel and are attached to the upper shell plates of the RPV. The RPV closure studs are made of A-540 Grade 23 and A-540 Grade 24 Class III. [Reference 2, Attachment 4]

The internal environment of the RPVs and CEDMs is that of the RCS, which contains water at an operating pressure of approximately 2250 psia. Normal RCS operating temperatures are approximately 548°F in the cold leg and 599.4°F in the hot leg. The RCS maintains a flow rate of approximately $134 \times 10^6$ lbm/hr. [Reference 1, Section 4.1.1, Table 4-1]. The RCS also contains chemicals for controlling reactor power and corrosion control. The RPVs and CEDMs/Electrical System is subjected to cyclic heat-up and cool-down during normal plant operation. The vessel closure head flange studs are placed under cyclic mechanical loading during vessel head attachment/removal and RCS heat-up and pressurization.

### Group 3 (low-cycle fatigue) - Aging Mechanism Effects

Low-cycle fatigue is the process of progressive localized permanent structural change occurring in a material subjected to conditions that produce fluctuating stresses and strains at some point or points, and which may culminate in cracks or complete fracture after a sufficient number of fluctuations. The low-cycle fatigue life of a component is the number of cycles of stress or strain that it experiences before fatigue failure. A component subjected to sufficient cycling with significant strain rates accumulates fatigue damage, which potentially can lead to crack initiation and crack growth. [Reference 24, page 4-7] The cracks may then propagate under continuing cyclic stresses.

The RPVs and CEDMs are subject to a wide variety of varying mechanical and thermal loads. Therefore, low-cycle fatigue is a plausible ARDM for all components in the scope of this section, except the RPV supports and snubber spacer blocks. [Reference 2, Attachments 5, 6, and 7] The RPV supports and snubber spacer blocks are not pressure boundary components, and are therefore not subject to the range of thermal and pressure stresses experienced by other RPV and CEDM components. Plant transients apply cyclical thermal loading and pressurization that contributes to low-cycle fatigue accumulation on the RPVs and CEDMs. The limiting locations for low-cycle fatigue are the RPV outlet coolant nozzles and RPV closure studs. The transient involving the greatest low-cycle fatigue accumulation for the RPV outlet coolant nozzles is the RCS cool-down from Mode 1 (i.e., full power) operation. For the RPV closure studs, the critical transient is the RCS heat-up. [Reference 25, Table 5-1]

Section III of the ASME B&PV Code (Winter 1967 Addenda) requires the design analysis for Class 1 vessels to address fatigue, and establishes limits such that initiation of fatigue cracks is precluded. Section III defines the threshold in terms of a cumulative usage factor (CUF). The low-cycle fatigue "damage" from a particular transient depends on the magnitude of the stresses applied. The summation of fatigue usage over all transients of all types is the CUF. Crack initiation is conservatively assumed to have occurred at a CUF equal to 1. [Reference 26]

The CUF can be determined from the actual transient history for the component and limits established on the number of transients. [Reference 27]

**Group 3 (low-cycle fatigue) - Methods to Manage Aging**

Mitigation: The effects of low-cycle fatigue can be mitigated by operational practices that reduce the number and severity of thermal transients on the RPV, and by proper design and material selection. For thick-walled components like the RPVs, the greatest membrane stresses due to thermal gradients and pressure occur during plant heat-up and cool-down. Significant stress cycles also occur from detensioning and retensioning RPV head studs for refueling. Therefore, the effects of low-cycle fatigue can be mitigated by operational practices that reduce the number and severity of pressure and thermal transients, by fuel management practices that minimize the number of refuelings, and by proper design and material selection.

Discovery: Fatigue cracks can be discovered by inspecting components, and the scope and frequency of inspections can be established based on the likelihood that fatigue cracks have initiated. As discussed above, low-cycle fatigue is accounted for in the original design in accordance with ASME Code Section III. Monitoring the number of design-basis transients and/or the accumulated fatigue usage can be used to predict the end of fatigue life.

The ASME Code Section III also provides accepted practices for analyzing Class I components for thermal fatigue combined with all other loads that must be considered under the CLB. An inspection program designed to identify crack initiation can be effective in discovering the effects of this aging mechanism prior to loss of the RCS pressure boundary function. The RPV closure studs, which are susceptible to mechanical (low cycle) fatigue, can be inspected during refueling outages when the RPV closure head is removed, and the RPV outlet coolant nozzles can be inspected during plant refueling.

**Group 3 (low-cycle fatigue) - Aging Management Program(s)**

Mitigation: As part of general operating practice, plant operators minimize the duration and severity of transitory operational cycles. Further modification of plant operating practices to reduce the magnitude and/or frequency of thermal transients would place additional unnecessary restrictions on plant operations. This is because the detection and monitoring activities discussed below are deemed adequate for effectively managing low-cycle fatigue in the RPV. No credit has been given to the 24-month fuel cycle since plant transients other than refueling could cause plant heat-ups and cool-downs.

Discovery: The CCNPP Fatigue Monitoring Program (FMP) records and tracks the number of critical thermal and pressure test transients. Cycle counting is performed as part of this program. The data for thermal transients is collected, recorded, and analyzed using FatiguePro software, which is a safety related software package. FatiguePro is used to analyze data that represents real transients. The FMP uses the results of FatiguePro to predict the number of transients for 40 and 60 years of plant operation. This information is used to verify that the RPV bounding locations will not experience more than 500 heat-up and cool-down cycles. [References 25, Tables 4-1 and 4-7; Reference 28] The Improved Technical Specifications for CCNPP, which will be implemented in 1997, will contain a requirement for tracking cyclic and transient occurrences to ensure that components are maintained within the design limits.

The current FMP monitors and tracks low-cycle fatigue usage for the selected components of the Nuclear Steam Supply System and the steam generators. Eleven locations in these systems have been selected for monitoring for low-cycle fatigue usage; they represent the most bounding locations for critical thermal and pressure transients, and operating cycles. [Reference 29] The RPV bounding locations for low-cycle fatigue are the RPV closure studs and RPV outlet coolant nozzles. The transient that has the greatest effect on fatigue life of the RPV coolant outlet nozzles are plant cool-down from Mode 1. For the closure head flange studs, the transient involving the greatest fatigue accumulation is plant heat-up. [Reference 25, Sections 4.1 and 4.8]

The design fatigue analysis (which was incorporated into the FMP) of the CCNPP RPVs determined the bounding locations and corresponding transients. Under the FMP, all other transients that contribute to low-cycle fatigue usage, including the RPV closure head flange stud tensioning, are accounted for as the initial fatigue usage. FatiguePro adds to subsequent fatigue usage resulting from RCS heat-up and cool-down transients to this "initial" fatigue usage to obtain the current CUF. [Reference 25, Sections 4.1 and 4.8; Reference 30]

The FMP tracks low-cycle fatigue usage using both cycle counting and stressed-based analysis. In accordance with ASME Code Section III, the fatigue life of a component is based on a calculated CUF of less than or equal to one. Cycle counting is used for the RPV outlet coolant nozzles and closure head flange studs, based on the original design transients in the ASME III, Class A design code analysis (Winter 1967 Addenda to the ASME B&PV Code). [Reference 1, Table 4-9; References 27, 28, 30]

Plant parameter data is collected on a periodic basis and reviewed to ensure that the data represents actual transients. Valid data is entered into FatiguePro, which counts the critical transient cycles and calculates the CUFs. Based on the ASME Code Section III, a CUF less than or equal to one, and/or the number of cycles remaining below the design allowable number, are acceptable conditions for any given component since no crack initiation would be predicted. The number of cycles and CUF are calculated on a semi-annual basis, which provides a readily predictable approach to the alert value. [Reference 27, Section 1.1] In order to stay within the design basis, corrective action is initiated well in advance of the CUF approaching one or the number of cycles approaching the design allowable, so that appropriate corrective actions can be taken in a timely and coordinated manner. [Reference 27]

The FMP will perform an engineering evaluation to determine if the low-cycle fatigue usage for the CEDM/RVLMS components are bounded by the existing bounding components. If they were not bounded, they will be added to the FMP. Tracking the usage for the limiting components ensures that all remaining components will also remain below their fatigue limits. [Reference 2, Attachment 2]

Modifications have been made to the FMP recognizing lessons learned. For example, analysis techniques, such as stress-based analysis, have been implemented for locations that have unique thermal transients or involve unique geometry. Other modifications have been made to reflect changes, or proposed changes, to plant operating practices to reflect plant operating conditions more accurately. The plant design change process requires the FMP to consider any proposed changes that affect the fatigue design basis or transient definitions. [References 28 and 32]

In conjunction with the FMP, the CCNPP ISI Program requirements, such as volumetric, visual, and/or surface examinations of Class I nuclear components, may discover unexpected crack initiation due to low-cycle fatigue. The requirements for specific components are outlined in ASME Section XI (Table IWB-2500) and flaw acceptance standards of IWB-3500, as implemented by the CCNPP ISI Program. [Reference 2, RPV, Attachment 6, Code O] Refer to the ISI Program discussion under Group 1 (general corrosion), Aging Management Programs.

Two CCNPP modifications have affected the RPV design basis loading. These involved the installation of permanent cavity seals and altering the stud tensioning sequence. The RPV design basis was reviewed for both of these modifications, and all fatigue loading changes were incorporated into the RPV analytical reports. The stud tensioning sequence modification did result in higher fatigue usage, but the analyzed end of life CUF is still below one. [Reference 32] As previously discussed, the FMP was modified to reflect these changes in plant operating conditions.

The CCNPP FMP has been inspected by the NRC, which noted that this monitoring system can be used to identify components where low-cycle fatigue usage may challenge the remaining and extended life of the components, and can provide a basis for corrective action where necessary. The program is controlled in accordance with the administrative procedures of the Life Cycle Management Program. [Reference 33] Since the FMP has been initiated, no locations have reached their design allowable number of cycles or a CUF of greater than or equal to one. The CUFs through 1996 for the RPV outlet coolant nozzles are 0.04537 (Unit 1) and 0.03349 (Unit 2). The CUF for the RPV closure studs is 0.28076 (Unit 1) and 0.25714 (Unit 2). [Reference 28]

To fully address low-cycle fatigue for license renewal, CCNPP has initiated an additional study, in conjunction with the Electric Power Research Institute, to evaluate the effects of low-cycle fatigue on various fatigue critical plant locations. The study will apply industry developed methodologies to identify fatigue sensitive component locations, which may require further evaluation or inspection for license renewal and evaluate environmental effects, as necessary. The program objective includes the development and justification of aging management practices for low-cycle fatigue at various component locations for the renewal period. [Reference 34]

Generic Safety Issue 166

Generic Safety Issue 166, Adequacy of Fatigue Life of Metal Components, presents concerns identified by the NRC which must be evaluated as part of the license renewal process. The NRC staff concerns about fatigue for license renewal fall into five categories: The first is adequacy of the fatigue design basis when environmental effects are considered. This concern does not apply to the RPV because of stringent RCS water chemistry controls, exceptionally low oxygen concentrations (less than five parts per billion), and because the RPV carbon steel interior surfaces are clad with stainless steel. The second category concerns the adequacy of both the number and severity of design-basis transients. Since these have already been analyzed for the CCNPP RPVs, this concern does not apply. A third category, adequacy of ISI requirements and procedures to detect fatigue indications, does not apply because CCNPP does not rely on ISI as the sole means for detection of fatigue. Category four, adequacy of the fatigue design basis for Class I piping components designed in accordance with ANSI B31.1, does not apply because the RPV and closure studs are designed in accordance with ASME Section III, Class I. The fifth and last category, adequacy of actions to be taken when the fatigue design basis is potentially compromised, as discussed above, are adequately addressed by the CCNPP FMP.

**Group 3 (low-cycle fatigue) - Demonstration of Aging Management**

Based on the factors presented above, the following conclusions can be reached with respect to the RPVs and CEDMs/Electrical System components subject to low-cycle fatigue:

- The RPV components have intended functions and their integrity must be maintained under CLB design conditions.
- Low-cycle fatigue is plausible for the entire RPV, except for the vessel support components and snubber spacer blocks.
- If left unmanaged, low-cycle fatigue could result in crack initiation and growth, which could impair the ability of the RPV to perform its intended function.
- The closure studs and outlet coolant nozzles are the bounding fatigue sensitive components for the RPV and are expected to bound the CEDM/RVLMS components.
- The CCNPP FMP tracks all applicable plant transients and monitors accumulated cycles and low-cycle fatigue usage for the bounding RPV components.
- The FMP is adequately controlled so that effective and timely corrective actions can be taken prior to a loss of RCS pressure boundary integrity resulting from fatigue damage.
- The FMP will be modified to include an engineering evaluation of the low-cycle fatigue usage for the CEDM/RVLMS components to ensure that their fatigue usage is bounded.
- Tracking the accumulated cycles and low-cycle fatigue usage for the bounding RPV components will ensure that all other RPV components will not exceed their fatigue design basis.
- ASME Section XI requirements provide for inspections that would discover cracks from any cause and require augmented inspections before fatigue crack initiation is expected.

Therefore, there is reasonable assurance that the effects of low-cycle fatigue in RPV components will be managed in order to maintain the components intended function under all design loading requirements of the CLB during the period of extended operation.

**Group 4 (neutron embrittlement) - Materials and Environment**

Table 4.2-2 shows that neutron embrittlement is plausible for only specific RPV components. This group includes the vessel plates and welds of the RPV lower shell, intermediate shell, and the lower portion of the nozzle shell courses. The RPV lower shell, intermediate shell, and the lower portion of the nozzle shell courses are fabricated from SA-533, Grade B, Class 1 low-alloy steel with an internal cladding of stainless steel. The associated welds are automatic submerged arc or manual metal arc with stainless steel cladding. [Reference 2, Attachments 4, 5, and 6]

The internal environment of the RPVs and CEDMs is that of the RCS, which contains chemically-treated, borated water at an operating pressure of approximately 2250 psia. Normal RCS operating temperatures are approximately 548°F in the cold leg and 599.4°F in the hot leg. The RCS flow rate during normal operation is approximately $134 \times 10^6$ lbm/hr. [Reference 1, Section 4.1.1, Table 4-1] The region of the RPV, where the components of this group are located, is an environment where neutron fluence is at its maximum levels with respect to components covered by this section.

The threshold for onset of neutron effects for RPV materials is conservatively defined to be a fast neutron fluence that exceeds 1E17 n/cm$^2$. [Reference 35, Appendix H] Portions of the RPV are expected to exceed this fluence. Certain welds in CCNPP Unit 1 are more sensitive to neutron exposure than was originally expected because of the amount of impurities (copper) in the Unit 1 RPV axial weld. [Reference 36]

**Group 4 (neutron embrittlement) - Aging Mechanism Effects**

Low alloy ferritic steels (like the RPV manganese-molybdenum-steel plates) that are exposed to a neutron fluence greater than 1E17 n/cm$^2$ are known to undergo microstructure changes that elevate the temperature at which the material begins to lose ductility (embrittlement) and may reduce its fracture toughness at normal operating temperatures (loss of upper shelf energy, LUSE). These effects depend on the level of certain alloying materials (nickel) and impurities (copper), and on the accumulated neutron exposure. [Reference 24, Section 4.1; Reference 37, Section 4.2.2] Understanding of the variables that cause these effects and their interdependencies continues to improve and is the subject of ongoing research by industry and NRC.

Therefore, neutron irradiation could reduce the fracture toughness of certain RPV materials, which in turn could reduce the ability of those materials to withstand temperature and pressure transients, including Pressurized Thermal Shock (PTS) transients. [Reference 24, page 5-3] Pressurized Thermal Shock transients are characterized as a severe rapid cool-down of the RPV, coincident with high or increasing pressure. The combined thermal and pressure stresses during temperature transients affecting low toughness materials increase the potential for extending flaws that may be present [Reference 37, page A-3-21] Therefore, neutron embrittlement (with accompanying LUSE) is considered a plausible ARDM for these components.

**Group 4 (neutron embrittlement) - Methods to Manage Aging**

Mitigation: The effects of neutron embrittlement and LUSE cannot be prevented, but can be mitigated by minimizing the neutron fluence to sensitive components, by using materials that are insensitive to these effects, and by reducing allowed system pressure during temperature transients and at reduced temperatures. Excessive neutron embrittlement and LUSE can be partially remedied by thermal annealing of the affected components.

Discovery: Practical methods to directly monitor neutron embrittlement of RPV components do not currently exist. However, this embrittlement can be monitored by periodic testing of coupons of representative RPV materials installed within the CCNPP RPVs, at another plant, or in test reactors. [Reference 35, Appendix H] Since the neutron exposure of the coupons is higher than that of the RPV, the embrittlement through the current license period and the period of extended operation can be predicted. These predictions can be adjusted to account for changes in fuel management, for the results of subsequent tests, and for subsequent research results.

**Group 4 (neutron embrittlement) - Aging Management Program(s)**

Mitigation: The current regulations require that neutron embrittlement and LUSE be managed. They establish specific embrittlement and LUSE limits; establish limits on pressure, temperature, and temperature transients; and require controls to minimize the potential for excessive stresses from pressure and temperature transients. [Reference 35, Section 61 and Appendix G] These mitigation elements are established at CCNPP by the LTOP controls, Pressure-Temperature limits, and neutron fluence limits included in Technical Specifications. No credit is taken for mitigation beyond those established in response to the regulatory requirements.

Discovery: The regulations require periodic testing of representative coupons of RPV materials to monitor neutron embrittlement and LUSE. [Reference 35, Section 61 and Appendix H] The CRVSP implements the requirements of 10 CFR Part 50, Appendix H, and provides the necessary data to monitor the embrittlement status of the reactor vessels. [Reference 1, Section 4.1.4.5] Calvert Cliffs has five surveillance capsules for each unit to provide sufficient RPV material property changes and fluence information as suggested in American Society for Testing and Materials (ASTM) E185-82 to meet the requirements of 10 CFR Part 50, Appendix H, through the current license period. Each CCNPP Unit also has one standby surveillance capsule to meet future needs (e.g., life extension, radical fuel management changes, etc.), as required. [Reference 38]

Because certain Unit 1 welds may be more susceptible to neutron embrittlement than originally expected, and because the RPV materials included in the original CCNPP surveillance program are less susceptible than the critical weld, BGE further extended this program into a CRVSP beginning in 1991. This CRVSP includes elements to identify and obtain test results and materials representative of the CCNPP RPVs from all available sources. The results of this ongoing program are chronicled extensively in the docketed submittals and responses as listed in References 39, 40, and 41. The results of this ongoing program include:

- Review of fabrication records for RPVs fabricated by CE to identify potential sources of information and archive material [Reference 39];
- Detailed review of CE fabrication records and industry databases to ensure all available data are properly considered [Reference 39];
- Incorporation of surveillance results from other plants that have surveillance coupons made using heats of weld wire identical to those in CCNPP beltlines [Reference 10, McGuire-1 Section];
- Acquisition of archival material and portions of the decommissioned reactor vessel from the Shoreham nuclear power plant made using heats of weld wire identical to those in CCNPP beltlines [Reference 10, Archive Section];
- Extensive chemical analysis of Shoreham and archival materials to assess chemical variability of RPV welds [Reference 39]; and
- Fabrication and installation of a supplemental Unit 1 surveillance capsule in 1988 containing the RPV material expected to be most affected by neutron irradiation. [Reference 10, Supplemental Surveillance Section]

These results to date demonstrate that CCNPP RPVs will remain well within established regulatory limits through the period of extended operation. NRC accepted these results as documented in the NRC correspondence listed in References 42, 43, 44, and 45.

This section of the BGE LRA is submitted to satisfy the requirements of 10 CFR 50.61(b)(1), which requires an updated assessment of the projected value of PTS "upon requests for a change in the expiration date for operation of the facility.". The NRC accepted BGE's results that CCNPP Units 1 and 2 beltline material are projected to be below PTS screening criteria 20 years after the original 40-year operating license. However, NRC noted that future chemistry and surveillance data may change their assessment. [Reference 42] Baltimore Gas and Electric Company recognized this possibility and purchased the rights to the surveillance capsule located in another pressurized water reactor. This capsule in another plant, as noted above, has provided for obtaining additional data for the Unit 1 material predicted by current regulations to be most susceptible to neutron irradiation effects. This data is expected to bound the neutron fluence of the subject RPV components through the period of extended operation, and to be available well before the period of extended operation. The CCNPP Unit 1 supplemental surveillance capsule also contains material identical to that material purchased above. Baltimore Gas and Electric Company's CRVSP provides for testing it in the next several years to provide further assurance that surveillance results from other plants reliably predict the behavior of CCNPP RPVs. [Reference 10, Supplemental Surveillance Section]

In addition, BGE is participating in CE Owners Group programs targeted toward improving the accuracy of current methods and industry standards for determining the resistance of RPV materials to initiation and propagation of cracks (fracture toughness). [Reference 46]

The regulations already require embrittlement and LUSE projections be updated to account for any significant changes in the projected values of $RT_{PTS}$ or change in the expiration date for operation of the facility. [Reference 35, Section 61] Baltimore Gas and Electric Company will continue to make periodic adjustments of neutron embrittlement and LUSE predictions, as needed, to account for any new information on the RPV beltline materials.

Therefore, there is reasonable assurance that the affects of neutron irradiation on the CCNPP RPVs will be known and future results will be addressed. Additional alternatives (e.g., annealing, further flux reduction, shielding) also remain that can be pursued if necessary.

**Group 4 (neutron embrittlement) - Demonstration of Aging Management**

Based on the factors presented above, the following conclusions can be reached with respect to the neutron embrittlement of RPV cylindrical shell plates and their associated welds:

- The vessel plates and welds of the RPV lower shell, intermediate shell, and the lower portion of the nozzle shell courses contribute to the RPV intended function, and their integrity must be maintained under CLB design conditions.
- These components are subject to significant neutron irradiation (fluence) due to their close proximity to the reactor core, and neutron embrittlement and LUSE are therefore plausible for the components. If not managed, these affects could result in sufficient loss of fracture

toughness to impair the ability of the RPV to perform its intended function under CLB design loading conditions.

- The CLB provides for monitoring the RPV materials for the effects of neutron irradiation, it provides specific limits on embrittlement and LUSE, it requires action before these limits are exceeded, and it provides specific constraints on operations to ensure allowable stresses are not exceeded.
- Baltimore Gas and Electric Company has demonstrated that the CCNPP RPVs will continue to meet CLB limits for embrittlement and LUSE for 60 years of operation, and has augmented its surveillance program to obtain embrittlement information that will bound the period of extended operation. NRC has concurred with this demonstration, noting that future test results may change this assessment.
- The CLB specifically requires that future test results for representative RPV materials be addressed and appropriate action taken such that RPV intended functions are assured.

Therefore, there is a reasonable assurance that the effects of neutron irradiation (embrittlement, LUSE) will be managed in order to maintain the RPV intended functions under all design conditions required by the CLB during the period of extended operation.

**Group 5 (stress corrosion cracking) - Materials and Environment**

Table 4.2-2 shows that SCC is plausible only for RPV support anchor bolts and for specific RPV components made of Alloys 600 and X-750. The RPV components susceptible to SCC and their material composition are the following: [Reference 2, Attachments 4, 5, and 6]

- RPV leakage monitoring tube (SB-167, SB-166);
- RPV ICI tube nozzle, Vent pipe, and CEDM nozzles (SB-167);
- RPV flow skirt (SB-168);
- RPV core stop lugs (SB-168);
- RPV core stabilizing lugs (SB-166 Alloys 600 and X-750);
- RPV surveillance capsule holders (SB-167, SB-167-65); and
- RPV anchor bolts (A-354 Grade B6).

The internal environment of the RPVs and CEDMs is that of the RCS, which contains chemically-treated water at an operating pressure of about 2250 psia. Normal RCS operating temperatures are approximately 548°F in the cold leg and 599.4°F in the hot leg. The normal RCS flow rate is approximately $134x10^6$ lbm/hr. [Reference 1, Section 4.1.1] The RPV anchor bolts are embedded in the concrete primary shield wall with the upper portions exposed to the containment environment. The containment environment is discussed in Group 1 (general corrosion).

As a result of the experience in 1989 and 1994 with minor Pressurizer Heater Sleeve leakage, BGE has replaced or scheduled near-term replacement of high-susceptibility Alloy 600 pressure boundary components. [Reference 6, Section 2]

The components identified as SB-166, 167, and 168 are susceptible to PWSCC. The following discussion on Alloy is also applicable to those components made of X-750.

**Group 5 (stress corrosion cracking) - Aging Mechanism Effects**

Alloy 600 materials and the RPV anchor bolts are susceptible to SCC, which occurs by the combined and synergistic interaction of a chemically-conducive environment, susceptible material, and tensile stress. Over long periods of time, Alloy 600 fails by slow, environmentally-induced crack initiation and growth, which may lead to eventual macroscopic plastic deformation. Understanding of the variables that cause these effects and their interdependencies continues to improve and is the subject of ongoing research by industry worldwide and by NRC. For the RPV anchor bolts, SCC would exhibit itself as continued cracking and eventual failure that could impair the supports' ability to withstand design basis loads.

Experience to date for Alloy 600 PWSCC indicates that for nozzles, cracks initiate first in the vicinity of penetrations (due to the higher residual stresses from welding the nozzles to the vessel/head), and then grow axially. The resulting cracks are short, grow slowly, grow at comparable rates axially and radially (through wall), and result in very minimal leakage when through-wall penetration finally occurs. For the portion of the nozzle external to the pressure boundary, crack growth that exhibits significant propagation around the circumference in a narrow axial region before penetrating through-wall would cause concern for separation and rapid development of significant leakage. [Reference 47]

Although safety concerns are minimal for Alloy 600 pressure boundary components, economic impacts can be significant since the ASME Code and NRC Regulations require the plant be shutdown for repair of any pressure boundary leakage. In addition, since circumferential cracking could lead to one or more design basis events (Loss of Coolant Accident, CEA Ejection), one cannot completely discount the potential for circumferential cracks that exhibit substantially greater circumferential growth rates than radial growth rates, based on the information presently available.

Therefore, the RPV components described above are considered susceptible to PWSCC, are exposed to an environment known to be conducive to PWSCC, and are placed under high tensile stresses. [Reference 2, Attachments 6 and 7] The combined effect of these factors could result in reduction of the ability of the components to maintain the RCS pressure boundary, to detect RCS leakage past the inner O-ring (leakage monitoring tube), to direct primary coolant flow through the core (flow skirt), to prevent excessive core displacement under specified accident conditions (core stop lugs), to limit flow induced vibrations in the core barrel (core stabilizing lugs), to support surveillance capsules (surveillance capsule holders), and to support the RPV position (RPV anchor bolts) under CLB design loading conditions. Therefore, SCC is a plausible ARDM for this group of components. [Reference 2, Attachments 4, 5, 6, and 7]

## Group 5 (stress corrosion cracking) - Methods to Manage Aging

Mitigation: The effects of SCC on Alloy 600 materials in the RCS cannot be eliminated, but can be monitored and actions taken to mitigate the effects. Sleeving, plating, weld overlays, thermal treatment, and replacement with material less susceptible to SCC can also be used to mitigate or remedy the effects of SCC. There are no methods to mitigate SCC on RPV anchor bolts. The effects of SCC could not be mitigated further for the RPVs and CEDMs/Electrical System beyond the strict chemistry control already used in the RCS. See the discussion of the RCS chemistry control program in Section 4.1 of the BGE LRA.

Discovery: Stress corrosion cracking of Alloy 600 components and RPV anchor bolts can be discovered and monitored by inspection programs. Inspection methods and frequencies can be defined based on susceptibility of the components, and inspection results from other facilities can be used to adjust the predicted susceptibility, inspection methods, and frequency of inspection.

Given the expected axial nature of Alloy 600 nozzle PWSCC cracks, the slow growth rates, the minimal leakage that occurs once through-wall penetration does occur, and the low safety concern, periodic inspections of low-susceptibility pressure boundary penetrations for evidence of leakage are sufficient. Dedicated inspection of high-susceptibility pressure boundary and non-pressure boundary components could be conducted and be timed based on expected initiation of cracks and expected propagation rates. This technique would have a high probability of discovering SCC effects prior to the loss of intended function.

Detection of PWSCC cracks shortly after they have initiated would permit timely repair, long before the intended function is jeopardized, and might minimize the cost and complexity of repair. Ranking models could be used to estimate PWSCC susceptibility and to schedule inspections based on the potential for crack initiation.

## Group 5 (stress corrosion cracking) - Aging Management Program(s)

The CCNPP Alloy 600 Program Plan was developed in response to primary pressure boundary leakage at CCNPP and other plants caused by PWSCC. The CCNPP Alloy 600 Program Plan builds on CCNPP and industry experience and provides for systematic evaluation of Alloy 600 pressure boundary components in the RCS, including the RPV and Pressurizer. It addresses nuclear safety concerns and identifies actions to minimize the safety and economic impact of SCC of Alloy 600 components. The program defines mitigation and discovery alternatives, as discussed below, and provides the process for considering susceptibility, safety, and economics in selecting from these alternatives. It also includes measures for monitoring industry experience and making appropriate adjustments based on this experience.

The susceptibility to PWSCC was evaluated for each CCNPP Alloy 600 nozzle based on ranking models developed by both Westinghouse and CE. A susceptibility index calculated from the Westinghouse model is a function of microstructure, effective stress factor, and temperature factor. The susceptibility index is used to develop a Relative Susceptibility Index, which is the susceptibility index of the component under analysis, as compared to the susceptibility index of the reference/benchmark component. The reference components in this case are the CCNPP Unit 2 Pressurizer heater sleeves, which developed minor leakage in 1989. The Relative Susceptibility Index is then multiplied by the actual or effective full power hours to obtain a time-dependent Relative Cumulative Susceptibility Index.

The CE model was used for the RCS Alloy 600 nozzles with inputs that were generic to all welded-tube type Alloy 600 nozzles; temperature, time in effective full power hours, and applied stress, which is based on the geometry of penetration and material yield strength. The CE model was used to calculate crack initiation probabilities as a function of effective full power hours. [Reference 6, Section 7].

The calculated susceptibility and crack initiation probability results were used to rank the nozzles and to develop recommendations for inspection, mitigation, repair and/or replacement of the nozzle(s). [Reference 6, Section 8] The susceptibility and economic analyses are used to select from the following options available for nozzles: [Reference 6, Section 9]

- Repair/replace nozzles based on susceptibility assessment.
- Perform mitigating techniques based on susceptibility assessment.
- Continue visually inspecting each nozzle as required by the boric acid corrosion program.
- Be prepared to repair nozzles on an as-failed basis. This option requires BGE to have replacement nozzles, repair plans, and design packages ready prior to the discovery of leakage.
- Perform augmented inspection to find non-throughwall PWSCC and perform repair/replacement as necessary.

Nuclear safety, ALARA (as low as reasonably achievable), and economics, are considered when selecting mitigating steps or repair/replacement for nozzles susceptible to PWSCC. Nuclear safety considerations include whether a complete severance of the nozzle due to circumferential cracking could lead to an unisolable small break loss-of-coolant accident, whether stresses would exist that could lead to such circumferential cracking, and whether a nozzle would exhibit minor leakage before crack growth would cause rapidly increasing leakage. [Reference 6, Section 14].

The focus of this program to date has been on pressure boundary components. This is appropriate given their greater stresses and greater potential to initiate design basis events. This program plan will be modified to include all Alloy-600 components in the RCS and RPVs, in addition to those that form the pressure boundary. [Reference 2, Attachments 2 and 10].

Mitigation: No program has been credited with mitigating the effects of SCC on RPV anchor bolting.

The CCNPP Alloy 600 Program Plan provides additional mitigation alternatives that include the following techniques: [Reference 6, Section 11]

- Shot peening - This induces compressive residual stress, slowing PWSCC initiation.
- Sleeving - A sleeve of Alloy 690 (less susceptible to PWSCC) is rolled and/or welded in existing Alloy 600 sleeves.
- Weld overlay - A thin layer of welded metal with a composition equivalent to Alloy 690 is deposited over the high stress area of the Alloy 600.
- Nickel plating - This technique provides a barrier to the primary water.

- Thermal treatment - Conducted in-situ to reduce residual stress.
- RCS temperature reduction - Reduces the thermodynamic driving force for PWSCC.
- Zinc Injection - Zinc added to the primary water may slow initiation and growth of PWSCC cracks.
- Mechanical stress improvement - controlled plastic deformation of the nozzle(s) in a manner which creates compressive residual stresses at locations susceptible to SCC (the technique has been used extensively in Boiling Water Reactor plants on stainless steel pipe fittings, weldments, and nozzles).

If mitigation techniques are not sufficient, then corrective actions are provided for nozzle(s) repair or replacement. The Alloy 600 Program Plan includes the following options to repair or replace nozzles: [Reference 6, Section 12]

- Local weld repair of defects;
- Replacement with Alloy 690 sleeves;
- Removal from service/plugging of a nozzle; or
- Encapsulate the existing nozzle in an outer nozzle bolted to the vessel to convert the nozzle into a bolted gasketed joint.

Discovery: The CCNPP ISI Program is credited with discovering SCC on the RPV anchor bolts. These RPV anchor bolts will be visually examined as defined in IWF-2500 of the ASME Code Section XI, and acceptance criteria contained in IWF-3410 of the ISI Program previously described in the Aging Management Program(s) section under Group 1 (general corrosion). [Reference 2, Attachment 6, Code S] The Boric Acid Corrosion Inspection Program augments the ISI Program for the discovery of SCC on the RPV anchor bolts. Refer to the Aging Management Program(s) in Group 1 (general corrosion) for a description of the Boric Acid Corrosion Inspection Program.

All RCS Alloy 600 nozzles are visually inspected each refueling outage for indications of leakage by the Boric Acid Corrosion Inspection Program. [Reference 18] Leakage that develops between refueling outages will be detected before significant through-wall leakage develops as a result of the Technical Specification limits on leakage. The Alloy 600 Program Plan also includes provisions for augmented inspection based on susceptibility.

RCS nozzles are evaluated under the Alloy 600 Program Plan based on primary and secondary factors. The primary evaluation factors for PWSCC susceptibility include: [Reference 6, Section 8]

- Operating temperature;
- Material peak stress level;
- Material heat treatment, if known;
- Number of Effective Full Power Hours; and
- Previous industry failures of same material heat.

The secondary evaluation factors for PWSCC susceptibility include: [Reference 6, Section 8]

- Industry susceptibility rankings;
- Amount and type of machining/rework on a component during fabrication;
- Product form (i.e., bar, tubing, pipe);
- Whether a crevice environment exists;
- Potential for trapping contaminants due to isolation from flow circulation (stagnation);
- History of chemical excursions; and
- General susceptibility of nozzle type.

Susceptibility rankings based on predictive models cannot be used to predict the exact timing of crack initiation or progression through-wall. Primary water stress corrosion cracking initiation times for identical materials vary over a wide band, and predictive models take into account a limited number of parameters. Detailed study of material properties, fabrication, and service history is required to assess susceptibility of individual nozzles. However, the susceptibility models are used to allow susceptibility comparison. The CE model is used in the economic analysis to determine the optimal time for augmented inspections, but not as the basis for safety evaluations. [Reference 6, Section 7]

The cracking initiation probability from the CE Model and the Relative Cumulative Susceptibility Index results are used for analyzing nozzles to determine when to perform augmented inspections for crack initiation. Alternatives for augmented nozzle inspections include eddy current, dye penetrant, and ultrasonic examination. [Reference 6, Section 10]

Relevant operating experience applicable to PWSCC includes failure of purification system resin retention screens. These resulted in a Unit 1 resin intrusion in March 1989, and a Unit 2 resin intrusion in January 1983. Resin decomposition products may contribute to cracking of sensitized Alloy 600, and an evaluation of the 1989 event by CE and BGE concluded that the increase in susceptibility to PWSCC was insignificant. [Reference 7]

Alloy 600 PWSCC has occurred at CCNPP and at other domestic and foreign pressurized water reactors, and BGE has been a leader in industry efforts to understand and manage PWSCC. [Reference 6, Section 3] The Alloy 600 Program Plan is a relatively new program, having been initiated in 1992. Since this program achieved its present form in 1995, no pressure boundary leakage has occurred as a result of PWSCC.

The Alloy 600 Program Plan includes specific provisions for monitoring industry experience and adjusting the plan accordingly. MN-3-304, Control of the Alloy 600 Program Plan, establishes administrative controls for this program under the site procedures hierarchy. The Alloy 600 Program Plan will continue to examine pressure boundary components susceptible to PWSCC to ensure that these components maintain their intended function required by the CLB during the period of extended operation. The program will be modified to include all Alloy 600 components that require AMR, including the above RPV components.

**Group 5 (stress corrosion cracking) - Demonstration of Aging Management**

Based on the factors presented above, the following conclusions can be reached with respect to SCC of the RPVs and CEDMs/Electrical System components:

- The RPV anchor bolts maintain the RPV position and are subject to SCC and the supports' integrity must be maintained under CLB design loading conditions.
- The components subject to PWSCC provide the RCS pressure-retaining boundary, monitor RCS leakage (leakage monitoring tube) into and from the space between the vessel head O-rings, reduce core inlet flow inequalities and prevent formation of large vortices (flow skirt), prevent excessive core displacement under specified accident conditions (core stop lugs), limit flow induced vibrations in the core barrel (core stabilizing lugs), and support surveillance capsules. Their integrity must be maintained under CLB design conditions.
- Although the susceptibility of pressure boundary components to PWSCC is low relative to most other plants, SCC is plausible for the components mentioned above, and could impair their ability to perform their intended functions.
- The CCNPP Alloy 600 Program Plan provides for actions to assess susceptibility and take action to mitigate, inspect, repair, or replace based on the results. It schedules augmented inspections when crack initiation is predicted to be more likely.
- The CCNPP Alloy 600 Program Plan includes provisions for monitoring and incorporating industry experience.
- The Alloy 600 Program Plan will be modified to include all applicable Alloy-600 components in the RCS, in addition to the RCS pressure boundary components.
- The CCNPP ISI Program, per the requirements of ASME Section XI, and Boric Acid Corrosion Inspection Program, provide for examination of RPV anchor bolts. Though SCC cannot be completely prevented, the status of the components can be evaluated on a regular basis and corrective actions can be taken as conditions indicate SCC.

Therefore, there is reasonable assurance that the effects of SCC and PWSCC will be managed in order to maintain the RPV intended functions under all conditions required by the CLB during the period of extended operation.

### 4.2.3 Conclusion

The programs discussed for the RPVs and CEDMs are listed on the following table. These programs are administratively controlled by a formal review and approval process. As demonstrated above, these programs will manage the aging mechanisms and their effects such that the intended functions of the RPVs and CEDMs will be maintained, consistent with the CLB during periods of extended operation.

The analysis/assessment, corrective action, and confirmation/documentation process for license renewal is in accordance with QL-2, "Corrective Actions Program." QL-2 is pursuant to 10 CFR Part 50, Appendix B, and covers all structures and components subject to an AMR.

TABLE 4.2-3

LIST OF AGING MANAGEMENT PROGRAMS FOR THE RPVs AND CEDMs / ELECTRICAL SYSTEM

| | Program | Credited As |
|---|---|---|
| Existing | CCNPP RV-22 RPV O-Ring Replacement | Discovery of general corrosion (Group 1) on the RPV head and vessel. |
| Existing | CCNPP MN-3-301, Boric Acid Corrosion Inspection Program | Discovery and mitigation of the effects of general corrosion (Group 1), discovery of mechanical wear (Group 2), and discovery of PWSCC on Alloy 600 nozzles (Group 5). |
| Existing | CCNPP RV-85, ICI Flange Cleaning and Inspection | Discovery of wear (Group 2) on the ICI tube nozzle flanges and associated components. |
| Existing | CCNPP RV-62 RPV, Stud, Nut, and Washer Cleaning | Discovery of wear (Group 2) and general corrosion (Group 1) on the RPV studs, nuts and washers. |
| Existing | CCNPP CRVSP | The CRVSP implements and augments the requirements of 10 CFR 50.61, Appendices G and H, to monitor the effects of neutron embrittlement (Group 4) of the RPV. |
| Existing | CCNPP ISI Program | Discovery, per ASME XI, and management of the effects of mechanical wear (Group 2), general corrosion (Group 1), and SCC (RPV anchor bolts in Group 6) on those RPV components susceptible to these ARDMs. |
| Modified | CCNPP RVLMS-2, Installation of the Flexible HJTC in the Reactor | Discovery of wear (Group 2) on the RVLMS flanges and associated components. RVLMS-2 will be modified to perform visual inspections of the Grayloc clamps, studs, nuts; and HJTC seal plug and drive nut for wear each time the RVLMS housing is reassembled. |
| Modified | CCNPP EN-1-300 FMP Procedure "Implementation of Fatigue Monitoring" | Discovery and management of the effects of low-cycle fatigue (Group 3). The FMP will be modified to perform an engineering evaluation for CEDM/RVLMS components to ensure that the components are bounded. |
| Modified | CCNPP Alloy 600 Program | Discovery and mitigation of the effects of PWSCC (Alloy 600 and Alloy X-750 materials in Group 5) on susceptible components. The Alloy 600 Program will be modified to include all Alloy-600 components, not just those which form the pressure boundary. |

### 4.2.4 References

1. CCNPP Updated Final Safety Analysis Report, Revision 19, April 1, 1996
2. "Reactor Vessel and Control Element Drive Mechanism Aging Management Review," Revision 1, April 11, 1997
3. Letter from Mr. J. T. Wiggins (NRC) to Mr. G. C. Creel (BGE), dated August 28, 1989, "NRC Region I Combined Inspection Report Nos. 50-317/89-14 and 50-318/89-14"
4. CE Report CENC-1849, "Evaluation of Calvert Cliffs Reactor Vessel Potential Wear of Bottom Head Clad Due to Loose Pump Bolt," December 16, 1988
5. CE Letter CSE-92-262, "Evaluation of Reactor Vessel Closure Head Stud Elongation for BGE Calvert Cliffs Unit 1," CR-9417-CSE92-1108, Revision 0, July 1, 1992
6. CCNPP Alloy 600 Program Plan, Revision 1, November 1996
7. Letter from Mr. J. T. Wiggins (NRC) to Mr. G. C. Creel (BGE), dated July 21, 1989, "NRC Region I Combined Inspection Report Nos. 50-317/89-06 and 50-318/89-06"
8. Southwest Research Institute, "The Preliminary Ultrasonic Examination of Weld Seams in the Calvert Cliffs Unit 1 Nuclear Reactor Vessel"
9. Southwest Research Institute, "1986 Inservice Examination of Selected Class 1 and Class 2 Components and Systems of Calvert Cliffs Nuclear Power Plant, Unit 1," April 1987
10. CCNPP Comprehensive Reactor Vessel Surveillance Program, Revision 2, June 1996
11. "CCNPP Low Temperature Overpressure Protection (LTOP) Controls Report," Revision 0, November 1995
12. CCNPP Procedure EN-1-214, "Low Temperature Overpressure Protection," Revision 0, March 30, 1995
13. CCNPP Component Level Scoping Results, Co[illegible] Rod Drive Mechanism and Electrical System, Revision 2, May 1995
14. CCNPP Procedure MN-3, "Pressure Boundary Codes and Special Processes Program," Revision 1
15. Inservice Inspection Program Plan for the Second Inspection Interval for Calvert Cliffs Nuclear Power Plant Units 1 & 2, Southwest Research Institute Project 17-1168, November 1987, Revision 0, Change 6, November 20, 1996
16. CCNPP Procedure MN-3-110 "Inservice Inspection of ASME Section XI Components," Revision 2, July 2, 1996
17. ASME Boiler and Pressure Vessel Code, Section XI, "Rules for In-Service Inspection of Nuclear Power Plant Components," 1983 Edition with Addenda through Summer 1983
18. CCNPP Procedure MN-3-301, "CCNPP Boric Acid Corrosion Inspection Program," Revision 1, December 15, 1994
19. CCNPP Procedure QL-2-100, "Issue Reporting and Assessment," Revision 4, January 2, 1996

20. CCNPP Procedure RV-62, "Reactor Vessel Stud, Nut and Washer Cleaning and Inspection," Revision 14, December 23, 1996

21. CCNPP Procedure RV-22, "Reactor Vessel Head O-Ring Replacement," Revision 12, October 24, 1996

22. Letter from Mr. C. H. Cruse (BGE) to NRC Document Control Desk, dated July 29, 1994, Licensee Event Report 94-004, Revision 01, "Excessive Corrosion of Incore Instrumentation Flange Components"

23. CCNPP Procedure RV-85, "Reactor Vessel ICI Flange Cleaning and Inspection," Revision 2, October 4, 1995

24. Electric Power Research Institute, "PWR Reactor Pressure Vessel License Renewal Industry Report, Revision 1," July 199436

25. CCNPP Fatigue Monitoring Program, Volumes 1 and 2, CE-NPSD-634-P, April 1992

26. ASME Boiler and Pressure Vessel Code, Section III, "Rules for Construction of Nuclear Vessels," 1965 Edition with Addenda through Winter 1967

27. CCNPP Administrative Procedure EN-1-300, "Implementation of Fatigue Monitoring," Revision 0, February 28, 1996

28. "CCNPP Fatigue Monitoring Report for 1996," Final Report for 1996 generated by CCNPP Administrative Procedure EN-1-300, "Implementation of Fatigue Monitoring"

29. CE Owners Group Report CE-NPSD-634-P, "Fatigue Monitoring Program for CCNPP Units 1 & 2," April 1, 1996

30. Structural Integrity Associates, Inc., SIR-96-006, "Cycle Counting and Cycle-Based Fatigue Report for CCNPP Units 1 and 2," February 21, 1996

31. CCNPP Engineering Standard ES-020, "Specialty Input Screens for the Engineering Service Process," Revision 1, May 1, 1996

32. CCNPP Engineering Service Package, ES199600182-000, "Revise Technical Manual 12017-074 for Tensioning and Detensioning Sequence of RPV Head Studs," Revision 0, February 13, 1996

33. Letter from Mr. J. P. Durr (NRC) to Mr. C. Stoiber (sic) (BGE), dated February 11, 1993, "Inspection Report Nos. 50-317/92-32 and 50-318/92-32"

34. BGE Procurement Specification No. 6422284S, "Technical Services to Evaluate Thermal Fatigue Effects on CCNPP Systems Requiring Aging Management Review for License Renewal," Revision 0, July 29, 1996

35. Title 10 of the Code of Federal Regulations (CFR) Part 50

36. Letter from Mr. G. C. Creel (BGE) to NRC Document Control Desk, dated December 1[illegible], 1991, "CCNPP Unit Nos. 1 and 2 50-317 & 50-318, Response to the 1991 Pressurized Thermal Shock Rule"

37. Electric Power Research Institute, "CCNPP Life Cycle Management/License Renewal Program-Reactor Pressure Vessel Evaluation," April 1995

38. Letter from Mr. C. H. Cruse (BGE) to NRC Document Control Desk, dated June 20, 1997, "Resubmittal of Request for Approval: Revision to Reactor Vessel Surveillance Capsule Withdrawal Schedule"

39. Letter from Mr. G. C. Creel (BGE) to NRC Document Control Desk, dated July 21, 1995, Request for Approval of Updated Values of Pressurized Thermal Shock (PTS) Reference Temperatures ($RT_{PTS}$) Values (10 CFR 50.61)

40. Letter from Mr. R. E. Denton (BGE) to NRC Document Control Desk, dated November 29, 1993, "Request to Use Plant-Specific Data for Reactor Vessel Fracture Toughness Analysis"

41. Letter from Mr. R. E. Denton (BGE) to NRC Document Control Desk, dated July 24, 1995, "Response to Generic Letter 92-01, Revision 1, Supplement 1: Reactor Vessel Integrity"

42. Letter from Mr. D. G. McDonald, Jr. (NRC) to Mr. R. E. Denton (BGE), dated January 2, 1996, "Updated Values for Pressurized Thermal Shock Reference Temperatures - Calvert Cliffs Nuclear Power Plant Unit Nos. 1 and 2," and attached NRC Safety Evaluation Report Pressurized Thermal Shock Evaluation

43. Letter from Mr. M. L. Boyle (NRC) to Mr. R. E. Denton (BGE), dated July 29, 1994, "NRC Safety Evaluation Report (SR) for BGE Request for Approval to Use Plant-Specific Data for Reactor Vessel Fracture Toughness Analysis, CCNPP Unit 1"

44. Letter from Mr. A. W. Dromerick (NRC) to Mr. C. H. Cruse (BGE), dated August 1, 1996, "Closeout for BGE Response to Generic Letter 92-01, Revision 1, Supplement 1, for the Calvert Cliffs Nuclear Power Plant, Unit Nos. 1 and 2"

45. NRC Memorandum from Mr. J. R. Strosnider to Mr. A. C. Thadani, dated May 5, 1995, "Assessment of Impact of Increased Variability in Chemistry on the $RT_{PTS}$ Value of PWR Reactor Vessels"

46. CEOG Final Report; CE-NPSD-1053, "Results of Fracture Toughness Testing of Unirradiated Linde 1092 RPV Welds - Phase 1," November 1996

47. NRC Generic Letter 97-01, "Degradation of Control Rod Drive Mechanism Nozzle and other Vessel Closure Head Penetrations," April 1, 1997