December 19, 1986

DMB 016

Docket Nos. 50-317
and 50-318

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Mr. J. A. Tiernan
Vice President - Nuclear Energy
Baltimore Gas & Electric Company
P. O. Box 1475
Baltimore, Maryland 21203

Dear Mr. Tiernan:

The Commission has issued the enclosed Amendment Nos. 125 and 106 to Facility Operating License Nos. DPR-53 and DPR-69 for Calvert Cliffs Nuclear Power Plant, Unit Nos. 1 and 2. These amendments consist of changes to the Technical Specifications (TS) in response to your application dated July 31, 1986, as supplemented by your letter dated November 5, 1986. Additionally, minor clarifying changes were made to these submittals with your staff's consent. The remaining items associated with your July 31, 1986 application will be addressed in future correspondence.

These amendments modify the Technical Specifications by (1) linking the completion of the reactor coolant pump (RCP) flywheel inspection required by TS surveillance 4.4.10.1.1 to the RCP motor overhaul program, and (2) making the administrative change prescribed by Generic Letter 84-13, "Technical Specification for Snubbers," by deleting the list of safety related hydraulic snubbers provided in Table 3.7-4 from the TS.

A copy of the related Safety Evaluation is also enclosed. The notice of issuance will be included in the Commission's next bi-weekly Federal Register notice.

Sincerely,

Original signed by

Scott A. Mc Neil, Project Manager
PWR Project Directorate #8
Division of PWR Licensing-B

Enclosures:
1. Amendment No. 125 to DPR-53
2. Amendment No. 106 to DPR-69
3. Safety Evaluation

cc w/enclosure:
See next page

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PDR ADOCK 05000317
P PDR

PBD#8	PBD#8	OGC-Bethesda	PBD#8
PMKreutzer	SMcNeil		AThadani
12/3/86	12/5/86	12/9/86	12/19/86

cc:
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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

BALTIMORE GAS AND ELECTRIC COMPANY

DOCKET NO. 50-317

CALVERT CLIFFS NUCLEAR POWER PLANT UNIT NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 125
License No. DPR-53

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

   A. The application for amendment by Baltimore Gas & Electric Company (the licensee) dated July 31, 1986, as supplemented by the November 5, 1986 submittal, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

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

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

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

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

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

   (2) Technical Specifications

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 125 , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of the date of its issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Ashok C. Thadani, Director
PWR Project Directorate #8
Division of PWR Licensing-B

Attachment:
Changes to the Technical
Specifications

Date of Issuance: December 19, 1986

ATTACHMENT TO LICENSE AMENDMENT NO. 125

FACILTIY OPERATING LICENSE NO. DPR-53

DOCKET NO. 50-317

Replace the following pages of the Appendix "A" Technical Specifications with the enclosed pages. The revised pages are identified by amendment number and contain vertical lines indicating the areas of change. The corresponding overleaf pages are provided to maintain document completeness.

Remove Pages	Insert Pages
3/4 4-28	3/4 4-28
3/4 7-25	3/4 7-25
3/4 7-26a	3/4 7-26a
3/4 7-26b	3/4 7-26b
3/4 7-27 through 3/4 7-62 inclusive	--
B 3/4 7-5	B 3/4 7-5
6-20	6-20

3/4.4.10 STRUCTURAL INTEGRITY

ASME CODE CLASS 1, 2 AND 3 COMPONENTS

LIMITING CONDITION FOR OPERATION

3.4.10.1 The structural integrity of ASME Code Class 1, 2 and 3 components shall be maintained in accordance with Specification 4.4.10.1.

APPLICABILITY: ALL MODES.

ACTION:

a. With the structural integrity of any ASME Code Class 1 component(s) not conforming to the above requirements, restore the structural integrity of the affected component(s) to within its limit or isolate the affected component(s) prior to increasing the Reactor Coolant System temperature more than 50°F above the minimum temperature required by NDT considerations.

b. With the structural integrity of any ASME Code Class 2 component(s) not conforming to the above requirements, restore the structural integrity of the affected component(s) to within its limit or isolate the affected component(s) prior to increasing the Reactor Coolant System temperature above 200°F.

c. With the structural integrity of any ASME Code Class 3 component(s) not conforming to the above requirements, restore the structural integrity of the affected component(s) to within its limit or isolate the affected component(s) from service.

d. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.4.10.1.1 The structural integrity of ASME Code Class 1, 2 and 3 components shall be demonstrated:

a. Per the requirements of Specification 4.0.5, and

b. Per the requirements of the augmented inservice inspection program specified in Specification 4.4.10.1.2.

REACTOR COOLANT SYSTEM

SURVEILLANCE REQUIREMENTS (Continued)

In addition to the requirements of Specification 4.0.5, each Reactor Coolant Pump flywheel shall be inspected per the recommendations of Regulatory Position C.4.b of Regulatory Guide 1.14, Revision 1, August 1975.*

4.4.10.1.2 Augmented Inservice Inspection Program for Main Steam and Main Feedwater Piping - The unencapsulated welds greater than 4 inches in nominal diameter in the main steam and main feedwater piping runs located outside the containment and traversing safety related areas or located in compartments adjoining safety related areas shall be inspected per the following augmented inservice inspection program using the applicable rules, acceptance criteria, and repair procedures of the ASME Boiler and Pressure Vessel Code, Section XI, 1974 Edition and Addenda through Summer 1975, for Class 2 components.

a. System integrity and baseline data shall be established by performing a 100% volumetric examination of each weld prior to exceeding 18 months of operation.

b. Each weld shall be examined in accordance with the above ASME Code requirements, except that 100% of the welds shall be examined, cumulatively, during each 10 year inspection interval. The welds to be examined during each inspection period shall be selected to provide a representative sample of the conditions of the welds. If these examinations reveal unacceptable structural defects in one or more welds, an additional 1/3 of the welds shall be examined and the inspection schedule for the repaired welds shall revert back to the first 10 year inspection program. If additional unacceptable defects are detected in the second sampling, the remainder of the welds shall also be inspected.

*Reactor coolant pump flywheel inspections for the first inservice inspection interval may be completed by June 1990 in conjunction with the reactor coolant pump motor overhaul program.

3/4.7.8 SNUBBERS

LIMITING CONDITION FOR OPERATION

3.7.8.1 All safety related snubbers[1] shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3 and 4. (MODES 5 and 6 for snubbers located on systems required OPERABLE in those MODES.

ACTION:

With one or more snubbers inoperable, within 72 hours replace or restore the inoperable snubber(s) to OPERABLE status, and perform an engineering evaluation* per Specification 4.7.8.b and c on the supporting component or declare the supported system inoperable and follow the appropriate ACTION statement for that system.

SURVEILLANCE REQUIREMENTS

4.7.8.1 Each snubber shall be demonstrated OPERABLE by performance of the following augmented inservice inspection program and the requirements of Specification 4.0.5. As used in this Specification, type of snubber shall mean snubbers of the same design and manufacturer, irrespective of capacity.

a. Visual inspections

Visual inspections shall be performed in accordance with the following schedule:

No. Inoperable Snubbers of Each Type per Inspection Period	Subsequent Visual** Inspection Period#
0	18 months $\pm$ 25%
1	12 months $\pm$ 25%
2	6 months $\pm$ 25%
3, 4	124 days $\pm$ 25%
5, 6, 7	62 days $\pm$ 25%
8 or more	31 days $\pm$ 25%

The snubbers may be further categorized into two groups: Those accessible and those inaccessible during reactor operation. Each group may be inspected independently in accordance with the above schedule.

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1 Safety related snubbers include those snubbers installed on safety related systems and snubbers on non-safety related systems if their failure or the failure of the system on which they are installed would have an adverse effect on any safety related system.

* A documented, visual inspection shall be sufficient to meet the requirements for an engineering evaluation. Additional analyses, as needed, shall be completed in a reasonable period of time.

** The inspection interval shall not be lengthened more than two steps at a time.

The provisions of Specification 4.0.2 are not applicable.

SURVEILLANCE REQUIREMENTS (Continued)

b. Visual Inspection Acceptance Criteria

Visual inspections shall verify (1) that there are no visible indications of damage or impaired OPERABILITY, and (2) that the snubber installation exhibits no visual indications of detachment from foundations or supporting structures. Snubbers which appear inoperable as a result of visual inspections may be determined OPERABLE for the purpose of establishing the next visual inspection interval, providing that (1) the cause of the rejection is clearly established and remedied for that particular snubber and for other snubbers that may be generically susceptible; and/or (2) the affected snubber is functionally tested in the as found condition and determined OPERABLE per Specification 4.7.8.d, as applicable. When the fluid port of a hydraulic snubber is found to be uncovered, the snubber shall be determined inoperable unless it can be determined OPERABLE via functional testing for the purpose of establishing the next visual inspection interval.

For the snubber(s) found inoperable, an engineering evaluation shall be performed on the component(s) which are supported by the snubber(s). The scope of this engineering evaluation shall be consistent with the licensee's engineering judgment and may be limited to a visual inspection of the supported component(s). The purpose of this engineering evaluation shall be to determine if the component(s) supported by the snubber(s) were adversely affected by the inoperability of the snubber(s) in order to ensure that the supported component remains capable of meeting the designed service.

c. Functional Tests

At least once per 18 months during shutdown, a representative sample of 10% of each type of snubbers in use in the plant shall be functionally tested either in place or in a bench test.* For each snubber that does not meet the functional test acceptance criteria of Specification 4.7.8.d, an additional 5% of that type snubber shall be functionally tested until no more failures are found or until all snubbers of that type have been functionally tested.

* The Steam Generator snubbers 1-63-13 through 1-63-28 need not be functionally tested until the refueling outage following June 30, 1985.

Snubbers identified as "Especially Difficult to Remove" or in "High Exposure Zones" shall also be included in the representative sample.*

In addition to the regular sample, snubbers which failed the previous functional test shall be retested during the next test period. If a spare snubber has been installed in place of a failed snubber, then both the failed snubber (if it is repaired and installed in another position) and the spare snubber shall be retested during the next test period. Failure of these snubbers shall not entail functional testing of additional snubbers.

If any snubber selected for functional testing either fails to lock up or fails to move, i.e., frozen in place, the cause will be evaluated and if caused by manufacturer or design deficiency all generically susceptible snubbers of the same design subject to the same defect shall be functionally tested. This testing requirement shall be independent of the requirements stated above for snubbers not meeting the functional test acceptance criteria.

For the snubber(s) found inoperable, an engineering evaluation shall be performed on the component(s) which are supported by the snubber(s). The scope of this engineering evaluation shall be consistent with the licensee's engineering judgment and may be limited to a visual inspection of the supported component(s). The purpose of this engineering evaluation shall be to determine if the component(s) supported by the snubber(s) were adversely affected by the inoperability of the snubber(s) in order to ensure that the supported component remains capable of meeting the designed service.

d. Hydraulic Snubbers Functional Test Acceptance Criteria

The hydraulic snubber functional test shall verify that:

1. Activation (restraining action) is achieved within the specified range of velocity or acceleration in both tension and compression.

2. Snubber bleed, or release rate, where required, is within the specified range in compression or tension. For snubbers specifically required to not displace under continuous load, the ability of the snubber to withstand load without displacement shall be verified.

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* Permanent or other exemptions from functional testing for individual snubbers in these categories may be granted by the Commission only if a justifiable basis for exemption is presented and/or snubber life destructive testing was performed to qualify snubber operability for all design conditions at either the completion of their fabrication or at a subsequent date.

SURVEILLANCE REQUIREMENTS (Continued)

e. Snubber Service Life Monitoring

A record of the service life of each snubber, the date at which the designated service life commences and the installation and maintenance records on which the designated service life is based shall be maintained as required by Specification 6.10.2.m.

At least once per 18 months, the installation and maintenance records for each safety related snubber shall be reviewed to verify that the indicated service life has not been exceeded or will not be exceeded prior to the next scheduled snubber service life review.* If the indicated service life will be exceeded prior to the next scheduled snubber service life review, the snubber service life shall be re-evaluated or the snubber shall be replaced or reconditioned so as to extend its service life beyond the date of the next scheduled service life review. This reevaluation, replacement, or reconditioning shall be indicated in the records.

* The provisions of Specification 4.0.2 are applicable.

PAGES 3/4 7-27 THROUGH 3/4 7-62 WERE DELETED BY AMENDMENT NO. ____.

environment. The operation of this system and the resultant effects on offsite dosage calculations was assumed in the accident analyses.

3/4.7.8 SNUBBERS

All safety related snubbers are required OPERABLE to ensure that the structural integrity of the reactor coolant system and all other safety related systems is maintained during and following a seismic or other event initiating dynamic loads. Snubbers excluded from this inspection program are those installed on non-safety related systems and then only if their failure or failure of the system on which they are installed would have no adverse effect on any safety-related system.

The visual inspection frequency is based upon maintaining a constant level of snubber protection to systems. Therefore, the required inspection interval varies inversely with the observed snubber failures and is determined by the number of inoperable snubbers of each type* found during an inspection. Inspections performed before that interval has elapsed may be used as a new reference point to determine the next inspection. However, the results of such early inspections performed before the original required time interval has elapsed (nominal time less 25%) may not be used to lengthen the required inspection interval. Any inspection whose results require a shorter inspection interval will override the previous schedule.

When the cause of the rejection of a snubber is clearly established and remedied for that snubber and for any other snubbers that may be generically susceptible, and verified by inservice functional testing, that snubber may be exempted from being counted as inoperable. Generically susceptible snubbers are those which are (1) of a specific make or model, (2) of the same design, and (3) similarly located or exposed to the same environmental conditions such as temperature, radiation, and vibration. These characteristics of the snubber installation shall be evaluated to determine if further functional testing of similar snubber installations is warranted.

When a snubber is found inoperable, an engineering evaluation is performed, in addition to the determination of the snubber mode of failure, in order to determine if any safety-related component or system has been adversely affected by the inoperability of the snubber. The engineering evaluation shall determine whether or not the snubber mode of failure has imparted a significant effect or degradation on the supported component or system.

To provide assurance of snubber functional reliability, a representative sample of the installed snubbers of each type* will be functionally tested during plant shutdowns at 18 month intervals. Observed failures of these sample snubbers shall require functional testing of additional units.

The service life of a snubber is evaluated via manufacturer input and information through consideration of the snubber service conditions and associated installation and maintenance records (newly installed snubber, seal replaced, spring replaced, in high radiation area, in high temperature area, etc....). The requirement to monitor the snubber service life is included to ensure that the

*Small bore (≤8") and large bore (>8") hydraulic snubbers are examples of different types of snubbers.

snubbers periodically undergo a performance evaluation in view of their age and operating conditions. The service life program is designed to uniquely reflect the conditions at Calvert Cliffs. The criteria for evaluating service life shall be determined, and documented, by the licensee. Records will provide statistical bases for future consideration of snubber service life. The requirements for the maintenance of records and the snubber service life review are not intended to affect plant operation.

3/4.7.9 SEALED SOURCE CONTAMINATION

The limitations on removable contamination for sources requiring leak testing, including alpha emitters, is based on 10 CFR 70.39(c) limits for plutonium. This limitation will ensure that leakage from byproduct, source, and special nuclear material sources will not exceed allowable intake values.

3/4.7.10 WATERTIGHT DOORS

This specification is provided to ensure the protection of safety related equipment from the effects of water or steam escaping from ruptured pipes or components in adjoining rooms.

3/4.7.11 FIRE SUPPRESSION SYSTEMS

The OPERABILITY of the fire suppression systems ensures that adequate fire suppression capability is available to confine and extinguish fires occurring in any portion of the facility where safety related equipment is located. The fire suppression system consists of the water system, spray and/or sprinklers, Halon and fire hose stations. The collective capability of the fire suppression systems is adequate to minimize potential damage to safety related equipment and is a major element in the facility fire protection program.

In the event that portions of the fire suppression systems are inoperable, alternate backup fire fighting equipment is required to be made available in the affected areas until the inoperable equipment is restored to service. Where a continuous fire watch is required in lieu of fire protection equipment and habitability due to heat or radiation is a concern, the fire watch should be stationed in a habitable area as close as possible to the inoperable equipment.

In the event the fire suppression water system becomes inoperable, immediate corrective measures must be taken since this system provides the major fire suppression capability of the plant. The requirement for a twenty-four hour report to the Commission provides for prompt evaluation of the acceptability of the corrective measures to provide adequate fire suppression capability for the continued protection of the nuclear plant.

6.10 RECORD RETENTION

6.10.1 The following records shall be retained for at least five years:

a. Records and logs of facility operation covering time interval at each power level.

b. Records and logs of principal maintenance activities, inspections, repair and replacement of principal items of equipment related to nuclear safety.

c. All REPORTABLE EVENTS.

d. Records of surveillance activities, inspections and calibrations required by these Technical Specifications.

e. Records of reactor tests and experiments.

f. Records of changes made to Operating Procedures.

g. Records of radioactive shipments.

h. Records of sealed source and fission detector leak tests and results.

i. Records of annual physical inventory of all sealed source material of record.

6.10.2 The following records shall be retained for the duration of the Facility Operating License:

a. Records and drawing changes reflecting facility design modifications made to systems and equipment described in the Final Safety Analysis Report.

b. Records of new and irradiated fuel inventory, fuel transfers and assembly burnup histories.

c. Records of facility radiation and contamination surveys.

d. Records of radiation exposure for all individuals entering radiation control areas.

e. Records of gaseous and liquid radioactive material released to the environs.

f. Records of transient or operational cycles for those facility components identified in Table 5.7-1.

g. Records of training and qualification for current members of the plant staff.

h. Records of in-service inspections performed pursuant to these Technical Specifications.

i. Records of Quality Assurance activities identified in the NRC approved QA Manual as lifetime records.

j. Records of reviews performed for changes made to procedures or equipment or reviews of tests and experiments pursuant to 10 CFR 50.59.

k. Records of meetings of the POSRC and the OSSRC.

l. Records of Environmental Qualification which are covered under the provisions of paragraph 6.13.

m. Records of the service lives of all safety related snubbers including the date at which the service life commences and associated installation and maintenance records.

6.11 RADIATION PROTECTION PROGRAM

Procedures for personnel radiation protection shall be prepared consistent with the requirements of 10 CFR Part 20 and shall be approved, maintained and adhered to for all operations involving personnel radiation exposure.

6.12 HIGH RADIATION AREA

6.12.1 In lieu of the "control device" or "alarm signal" required by paragraph 20.203(c)(2) of 10 CFR Part 20:

a. A high radiation area in which the intensity of radiation is greater than 100 mrem/hr but less than 1000 mrem/hr shall be barricaded and conspicuously posted as a High Radiation Area and entrance thereto shall be controlled by issuance of a Special or Radiation Work Permit and any individual or group of individuals permitted to enter such areas shall be provided with a radiation monitoring device which continuously indicates the radiation dose rate in the area.

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

BALTIMORE GAS AND ELECTRIC COMPANY

DOCKET NO. 50-318

CALVERT CLIFFS NUCLEAR POWER PLANT UNIT NO. 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 106
License No. DPR-69

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

   A. The application for amendment by Baltimore Gas & Electric Company (the licensee) dated July 31, 1986, as supplemented by the November 5, 1986 submittal, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

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

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

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

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

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

   2. <u>Technical Specifications</u>

      The Technical Specifications contained in Appendix A, as revised through Amendment No. 106 , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of the date of its issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

A C Thadani

Ashok C. Thadani, Director
PWR Project Directorate #8
Division of PWR Licensing-B

Attachment:
Changes to the Technical Specifications

Date of Issuance: December 19, 1986

ATTACHMENT TO LICENSE AMENDMENT NO. 106

FACILTIY OPERATING LICENSE NO. DPR-69

DOCKET NO. 50-318

Replace the following pages of the Appendix "A" Technical Specifications with the enclosed pages. The revised pages are identified by amendment number and contain vertical lines indicating the areas of change. The corresponding overleaf pages are provided to maintain document completeness.

Remove Pages	Insert Pages
3/4 4-29	3/4 4-29
3/4 7-25	3/4 7-25
3/4 7-26a	3/4 7-26a
3/4 7-26b	3/4 7-26b
3/4 7-27 through 3/4 7-54 inclusive	--
B 3/4 7-5	B 3/4 7-5
6-20	6-20

SURVEILLANCE REQUIREMENTS (Continued)

In addition to the requirements of Specification 4.0.5, each Reactor Coolant Pump flywheel shall be inspected per the recommendations of Regulatory Position C.4.b of Regulatory Guide 1.14, Revision 1, August 1975.*

4.4.10.1.2 Augmented Inservice Inspection Program for Main Steam and Main Feedwater Piping - The unencapsulated welds greater than 4 inches in nominal diameter in the main steam and main feedwater piping runs located outside the containment and traversing safety related areas or located in compartments adjoining safety related areas shall be inspected per the following augmented inservice inspection program using the applicable rules, acceptance criteria, and repair procedures of the ASME Boiler and Pressure Vessel Code, Section XI, 1974 Edition and Addenda through Summer 1975, for Class 2 components.

a. System integrity and baseline data shall be established by performing a 100% volumetric examination of each weld prior to exceeding 18 months of operation.

b. Each weld shall be examined in accordance with the above ASME Code requirements, except that 100% of the welds shall be examined, cumulatively, during each 10 year inspection interval. The welds to be examined during each inspection period shall be selected to provide a representative sample of the conditions of the welds. If these examinations reveal unacceptable structural defects in one or more welds, an additional 1/3 of the welds shall be examined and the inspection schedule for the repaired welds shall revert back to the first 10 year inspection program. If additional unacceptable defects are detected in the second sampling, the remainder of the welds shall also be inspected.

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*Reactor coolant pump flywheel inspections for the first inservice inspection interval may be completed by June 1991 in conjunction with the reactor coolant pump motor overhaul program.

REACTOR COOLANT SYSTEM

CORE BARREL MOVEMENT

LIMITING CONDITION FOR OPERATION

3.4.11 Core barrel movement shall be limited to less than the Amplitude Probability Distribution (APD) and Spectral Analysis (SA) Alert Levels for the applicable THERMAL POWER level.

APPLICABILITY: MODE 1.

ACTION:

a. With the APD and/or SA exceeding their applicable Alert Levels, POWER OPERATION may proceed provided the following actions are taken:

   1. APD shall be measured and processed at least once per 24 hours,

   2. SA shall be measured at least once per 24 hours and shall be processed at least once per 7 days, and

   3. A Special Report, identifying the cause(s) for exceeding the applicable Alert Level, shall be prepared and submitted to the Commission pursuant to Specification 6.9.2 within 30 days of detection.

b. With the APD and/or SA exceeding their applicable Action Levels, measure and process APD and SA data within 24 hours to determine if the core barrel motion is exceeding its limits. With the core barrel motion exceeding its limits, reduce the core barrel motion to within its Action Levels within the next 24 hours or be in HOT STANDBY within the following 6 hours.

c. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

PLANT SYSTEMS

3/4.7.8 SNUBBERS

LIMITING CONDITION FOR OPERATION

3.7.8.1 All safety related snubbers[1] shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3 and 4. (MODES 5 and 6 for snubbers located on systems required OPERABLE in those MODES.)

ACTION:

With one or more snubbers inoperable, within 72 hours replace or restore the inoperable snubber(s) to OPERABLE status, and perform an engineering evaluation* per Specification 4.7.8.b and c on the supported component or declare the supported system inoperable and follow the appropriate ACTION statement for that system.

SURVEILLANCE REQUIREMENTS

4.7.8.1 Each snubber shall be demonstrated OPERABLE by performance of the following augmented inservice inspection program and the requirements of Specification 4.0.5. As used in this Specification, type of snubber shall mean snubbers of the same design and manufacturer, irrespective of capacity.

a. Visual Inspections

Visual inspections shall be performed in accordance with the following schedule:

No. Inoperable Snubbers of Each Type per Inspection Period	Subsequent Visual** Inspection Period#
0	18 months ± 25%
1	12 months ± 25%
2	6 months ± 25%
3, 4	124 days ± 25%
5, 6, 7	62 days ± 25%
8 or more	31 days ± 25%

The snubbers may be further categorized into two groups: Those accessible and those inaccessible during reactor operation. Each group may be inspected independently in accordance with the above schedule.

---

1 Safety related snubbers include those snubbers installed on safety related systems and snubbers on non-safety related systems if their failure or the failure of the system on which they are installed would have an adverse effect on any safety related system.

* A documented, visual inspection shall be sufficient to meet the requirements for an engineering evaluation. Additional analyses, as needed, shall be completed in a reasonable period of time.

** The inspection interval shall not be lengthened more than two steps at a time.

\# The provisions of Specification 4.0.2 are not applicable.

SURVEILLANCE REQUIREMENTS (Continued)

b. Visual Inspection Acceptance Criteria

Visual inspections shall verify (1) that there are no visible indications of damage or impaired OPERABILITY, and (2) that the snubber installation exhibits no visual indications of detachment from foundations or supporting structures. Snubbers which appear inoperable as a result of visual inspections may be determined OPERABLE for the purpose of establishing the next visual inspection interval, providing that (1) the cause of the rejection is clearly established and remedied for that particular snubber and for other snubbers that may be generically susceptible; and/or (2) the affected snubber is functionally tested in the as found condition and determined OPERABLE per Specification 4.7.8.d, as applicable. When the fluid port of a hydraulic snubber is found to be uncovered, the snubber shall be determined inoperable unless it can be determined OPERABLE via functional testing for the purpose of establishing the next visual inspection interval.

For the snubber(s) found inoperable, an engineering evaluation shall be performed on the component(s) which are supported by the snubber(s). The scope of this engineering evaluation shall be consistent with the licensee's engineering judgment and may be limited to a visual inspection of the supported component(s). The purpose of this engineering evaluation shall be to determine if the component(s) supported by the snubber(s) were adversely affected by the inoperability of the snubber(s) in order to ensure that the supported component remains capable of meeting the designed service.

c. Functional Tests

At least once per 18 months during shutdown, a representative sample of 10% of each type of snubbers in use in the plant shall be functionally tested either in place or in a bench test.* For each snubber that does not meet the functional test acceptance criteria of Specification 4.7.8.d, an additional 5% of that type snubber shall be functionally tested until no more failures are found or until all snubbers of that type have been functionally tested.

* The Steam Generator snubbers 2-63-11 through 2-63-26 need not be functionally tested until the refueling outage following June 30, 1985.

Snubbers identified as "Especially Difficult to Remove" or in "High Radiation Zones" shall also be included in the representative sample.*

In addition to the regular sample, snubbers which failed the previous functional test shall be retested during the next test period. If a spare snubber has been installed in place of a failed snubber, then both the failed snubber (if it is repaired and installed in another position) and the spare snubber shall be retested during the next test period. Failure of these snubbers shall not entail functional testing of additional snubbers.

If any snubber selected for functional testing either fails to lock up or fails to move, i.e., frozen in place, the cause will be evaluated and if caused by manufacturer or design deficiency all generically susceptible snubbers of the same design subject to the same defect shall be functionally tested. This testing requirement shall be independent of the requirements stated above for snubbers not meeting the functional test acceptance criteria.

For the snubber(s) found inoperable, an engineering evaluation shall be performed on the component(s) which are supported by the snubber(s). The scope of this engineering evaluation shall be consistent with the licensee's engineering judgment and may be limited to a visual inspection of the supported component(s). The purpose of this engineering evaluation shall be to determine if the component(s) supported by the snubber(s) were adversely affected by the inoperability of the snubber(s) in order to ensure that the supported component remains capable of meeting the designed service.

d. Hydraulic Snubbers Functional Test Acceptance Criteria

The hydraulic snubber functional test shall verify that:

1. Activation (restraining action) is achieved within the specified range of velocity or acceleration in both tension and compression.

2. Snubber bleed, or release rate, where required, is within the specified range in compression or tension. For snubbers specifically required to not displace under continuous load, the ability of the snubber to withstand load without displacement shall be verified.

* Permanent or other exemptions from functional testing for individual snubbers in these categories may be granted by the Commission only if a justifiable basis for exemption is presented and/or snubber life destructive testing was performed to qualify snubber operability for all design conditions at either the completion of their fabrication or at a subsequent date.

SURVEILLANCE REQUIREMENTS (Continued)

e. Snubber Service Life Monitoring

A record of the service life of each snubber, the date at which the designated service life commences and the installation and maintenance records on which the designated service life is based shall be maintained as required by Specification 6.10.2.m.

At least once per 18 months, the installation and maintenance records for each safety related snubber shall be reviewed to verify that the indicated service life has not been exceeded or will not be exceeded prior to the next scheduled snubber service life review.* If the indicated service life will be exceeded prior to the next scheduled snubber service life review, the snubber service life shall be reevaluated or the snubber shall be replaced or reconditioned so as to extend its service life beyond the date of the next scheduled service life review. This reevaluation, replacement or reconditioning shall be indicated in the records.

* The provisions of Specification 4.0.2 are applicable.

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environment. The operation of this system and the resultant effects on offsite dosage calculations was assumed in the accident analyses.

3/4.7.8 SNUBBERS

All safety related snubbers are required OPERABLE to ensure that the structural integrity of the reactor coolant system and all other safety related systems is maintained during and following a seismic or other event initiating dynamic loads. Snubbers excluded from this inspection program are those installed on non-safety related systems and then only if their failure or failure of the system on which they are installed would have no adverse effect on any safety-related system.

The visual inspection frequency is based upon maintaining a constant level of snubber protection to systems. Therefore, the required inspection interval varies inversely with the observed snubber failures and is determined by the number of inoperable snubbers of each type* found during an inspection. Inspections performed before that interval has elapsed may be used as a new reference point to determine the next inspection. However, the results of such early inspections performed before the original required time interval has elapsed (nominal time less 25%) may not be used to lengthen the required inspection interval. Any inspection whose results require a shorter inspection interval will override the previous schedule.

When the cause of the rejection of a snubber is clearly established and remedied for that snubber and for any other snubbers that may be generically susceptible, and verified by inservice functional testing, that snubber may be exempted from being counted as inoperable. Generically susceptible snubbers are those which are (1) of a specific make or model, (2) of the same design, and (3) similarly located or exposed to the same environmental conditions such as temperature, radiation, and vibration. These characteristics of the snubber installation shall be evaluated to determine if further functional testing of similar snubber installations is warranted.

When a snubber is found inoperable, an engineering evaluation is performed, in addition to the determination of the snubber mode of failure, in order to determine if any safety-related component or system has been adversely affected by the inoperability of the snubber. The engineering evaluation shall determine whether or not the snubber mode of failure has imparted a significant effect or degradation on the supported component or system.

To provide assurance of snubber functional reliability, a representative sample of the installed snubbers of each type* will be functionally tested during plant shutdowns at 18 month intervals. Observed failures of these sample snubbers shall require functional testing of additional units.

The service life of a snubber is evaluated via manufacturer input and information through consideration of the snubber service conditions and associated installation and maintenance records (newly installed snubber, seal replaced, spring replaced, in high radiation area, in high temperature area, etc....). The requirement to monitor the snubber service life is included to ensure that the

*Small bore (≤8") and large bore (>8") hydraulic snubbers are examples of different types of snubbers.

snubbers periodically undergo a performance evaluation in view of their age and operating conditions. The service life program is designed to uniquely reflect the conditions at Calvert Cliffs. The criteria for evaluating service life shall be determined, and documented, by the licensee. Records will provide statistical bases for future consideration of snubber service life. The requirements for the maintenance of records and the snubber service life review are not intended to affect plant operation.

3/4.7.9 SEALED SOURCE CONTAMINATION

The limitations on removable contamination for sources requiring leak testing, including alpha emitters, is based on 10 CFR 70.39(c) limits for plutonium. This limitation will ensure that leakage from byproduct, source, and special nuclear material sources will not exceed allowable intake values.

3/4.7.10 WATERTIGHT DOORS

This specification is provided to ensure the protection of safety related equipment from the effects of water or steam escaping from ruptured pipes or components in adjoining rooms.

3/4.7.11 FIRE SUPPRESSION SYSTEMS

The OPERABILITY of the fire suppression systems ensures that adequate fire suppression capability is available to confine and extinguish fires occurring in any portion of the facility where safety related equipment is located. The fire suppression system consists of the water system, spray and/or sprinklers, Halon and fire hose stations. The collective capability of the fire suppression systems is adequate to minimize potential damage to safety related equipment and is a major element in the facility fire protection program.

In the event that portions of the fire suppression systems are inoperable, alternate backup fire fighting equipment is required to be made available in the affected areas until the inoperable equipment is restored to service. Where a continuous fire watch is required in lieu of fire protection equipment and habitability due to heat or radiation is a concern, the fire watch should be stationed in a habitable area as close as possible to the inoperable equipment.

In the event the fire suppression water system becomes inoperable, immediate corrective measures must be taken since this system provides the major fire suppression capability of the plant. The requirement for a twenty-four hour report to the Commission provides for prompt evaluation of the acceptability of the corrective measures to provide adequate fire suppression capability for the continued protection of the nuclear plant.

6.10 RECORD RETENTION

6.10.1 The following records shall be retained for at least five years:

a. Records and logs of facility operation covering time interval at each power level.

b. Records and logs of principal maintenance activities, inspections, repair and replacement of principal items of equipment related to nuclear safety.

c. All REPORTABLE EVENTS.

d. Records of surveillance activities, inspections and calibrations required by these Technical Specifications.

e. Records of reactor tests and experiments.

f. Records of changes made to Operating Procedures.

g. Records of radioactive shipments.

h. Records of sealed source and fission detector leak tests and results.

i. Records of annual physical inventory of all sealed source material of record.

6.10.2 The following records shall be retained for the duration of the Facility Operating License:

a. Records and drawing changes reflecting facility design modifications made to systems and equipment described in the Final Safety Analysis Report.

b. Records of new and irradiated fuel inventory, fuel transfers and assembly burnup histories.

c. Records of facility radiation and contamination surveys.

d. Records of radiation exposure for all individuals entering radiation control areas.

e. Records of gaseous and liquid radioactive material released to the environs.

f. Records of transient or operational cycles for those facility components identified in Table 5.7.1.

g. Records of training and qualification for current members of the plant staff.

h. Records of in-service inspections performed pursuant to these Technical Specifications.

i. Records of Quality Assurance activities identified in the NRC approved QA Manual as lifetime records.

j. Records of reviews performed for changes made to procedures or equipment or reviews of tests and experiments pursuant to 10 CFR 50.59.

k. Records of meetings of the POSRC and the OSSRC.

l. Records of Environmental Qualification which are covered under the provisions of paragraph 6.13.

m. Records of the service lifes of all safety related snubbers including the date at which the service life commences and associated installation and maintenance records.

6.11 RADIATION PROTECTION PROGRAM

Procedures for personnel radiation protection shall be prepared consistent with the requirements of 10 CFR Part 20 and shall be approved, maintained and adhered to for all operations involving personnel radiation exposure.

6.12 HIGH RADIATION AREA

6.12.1 In lieu of the "control device" or "alarm signal" required by paragraph 20.203(c)(2) of 10 CFR Part 20:

a. A high radiation area in which the intensity of radiation is greater than 100 mrem/hr but less than 1000 mrem/hr shall be barricaded and conspicuously posted as a High Radiation Area and entrance thereto shall be controlled by issuance of a Special or Radiation Work Permit and any individual or group of individuals permitted to enter such areas shall be provided with a radiation monitoring device which continuously indicates the radiation dose rate in the area.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NOS.125 AND 106

TO FACILITY OPERATING LICENSE NOS. DPR-53 AND DPR-69

BALTIMORE GAS AND ELECTRIC COMPANY

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

DOCKET NOS. 50-317 AND 50-318

INTRODUCTION

By the application for license amendments dated July 31, 1986, as supplemented by the November 5, 1986 submittal, the Baltimore Gas & Electric Company (BG&E) requested changes to the Technical Specifications (TS) for Calvert Cliffs, Units 1 and 2. The TS changes proposed are as follows:

1) Modify TS Surveillance Requirement 4.4.10.1.1 for Unit 1 (Unit 2) by adding the note:

   "Reactor coolant pump flywheel inspections for the first inservice inspection interval may be completed by June 1990 (June 1991) in conjunction with the reactor coolant pump motor overhaul program."

This note was paraphrased from the original BG&E amendment request for the purpose of clarification. These changes were made with the consent of the BG&E staff.

2) Modify TS 3/4.7.8, "Snubbers," by deleting Table 3.7-4, "Safety Related Hydraulic Snubbers," and by changing the TS Limiting Condition for Operation (LCO) 3.7.8.1 to state:

   "All safety related snubbers shall be operable", and all other references to "snubbers listed in Table 3.7-4 shall be appropriately changed to "safety related snubbers" to meet the guidelines of LCO 3.7.8.1.

In addition the following note shall be added to TS LCO 3.7.8.1:

   "Safety related snubbers include those snubbers installed on safety related systems and snubbers on non-safety related systems if their failure or the failure of the system on which they are installed would have an adverse effect on any safety related system."

DISCUSSION AND EVALUATION

Units 1 and 2 TS Surveillance Requirement 4.4.10.1.1 requires RCP flywheel inspections to be conducted per the recommendations of Regulatory Position C.4.b of Regulatory Guide 1.14, Revision 1, dated August 1975. Regulatory Position C.4.b.(2) states that a surface examination of all exposed surfaces and complete ultrasonic volumetric examination of the RCP flywheels shall be performed at approximately 10-year intervals during the plant shutdown coinciding with the inservice inspection schedule as required by Section XI of the ASME Code.

In Change No. 1, the licensee has proposed that this requirement be modified to link the performance of the RCP flywheel inspections to the licensee's voluntary RCP motor overhaul program rather than to the ISI interval. This proposed modification would only affect RCP flywheel inspections applicable to the first 10-year ISI interval. All following flywheel inspections would continue to be linked to their respective ISI interval schedules.

Three years ago the licensee initiated a voluntary RCP motor overhaul program utilizing a spare motor to enable quick motor changeout. During each refueling outage one motor will be changed until all eight RCP motors are overhauled, at which time the voluntary program will be completed. The licensee intends to perform the required flywheel examinations coincident with the motor overhaul schedule. To date, two RCP flywheels have been inspected with acceptable results. As a consequence of linking the RCP flywheel inspection to the RCP motor overhaul program, four of the eight RCP flywheels will have had their inspections completed at the end of the first 10-year ISI interval.

The first 10-year ISI intervals for both Units 1 and 2 are scheduled for completion in April 1987. This proposed change would result in the completion of the RCP flywheel inspections being deferred to June 1990, and June 1991 for Units 1 and 2, respectively, due to being linked to the completion of the RCP motor overhaul program.

Though this proposal would significantly lengthen the period of time necessary to complete the RCP flywheel inspections, the technical superiority of the visual flywheel inspection conducted in conjunction with the RCP motor changeout in comparison to the difficulty in adequately performing the conventional in-place ultrasonic examination of the two-piece bolted flywheel design more than compensates for the increase in inspection time.

Accordingly, the proposed change to TS 4.4.10.1.1 is deemed acceptable in that, though the period of time over which the inspection is performed is lengthened, the inspection results provided are better than for a conventional in-place ultrasonic examination.

The Change No. 2 proposal to delete Table 3.7-4, "Safety Related Hydraulic Snubbers," from the Units 1 and 2 TS 3/4.7.8, "Snubbers," and replace the phrase "snubbers listed in Table 3.7-4" with "safety-related snubbers" is an administrative change that is in accordance with the Commission guidelines presented in Generic Letter (GL) 84-13, "Technical Specifications for Snubbers."

GL 84-13 states that the Commission has reassessed the inclusion of snubber listings within the TS and has concluded that such listings are unnecessary provided the snubber TS are modified to specify which snubbers are required to be operable. The snubbers that were recommended for required operability by GL 84-13 included all snubbers with the exception of those snubbers installed on non-safety related systems whose failure or failure of the system on which they are installed would have no adverse effect on any safety related systems. The licensee's proposal that only safety related snubbers shall be operable complies with the guidance of GL 84-13 as the licensee's definition of safety related snubbers in the LCO includes all snubbers as specified by GL 84-13, and also includes all snubbers currently listed in Table 3.7-4.

This change is administrative in nature. Therefore, the proposed change to TS 3/4.7.8 to delete Table 3.7-4 is acceptable. Though the snubber table is being deleted, all of the snubbers that are currently required to be operable will still be required to be demonstrated operable in the proposed requirement.

ENVIRONMENTAL CONSIDERATION

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

CONCLUSION

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

Date: December 19, 1986

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