-DCR-016

June 17, 1986

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. 118 and 110 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 in partial response to your application dated January 20, 1986 as supplemented by your letter dated April 25, 1986. The remaining issues associated with this application will be addressed in future correspondence.

The amendments change the Unit 1 and Unit 2 Technical Specifications (TS) as follows: (1) TS 4.7.1.2c.2, "Auxiliary Feedwater System," is changed to reflect a revised flow test requirement. In addition, a typographical error is corrected (Unit 2 only); (2) TS 4.7.8.1, "Snubbers," is changed to subdivide the snubber surveillance population based upon snubber type; and (3) TS 4.9.12d.2, "Spent Fuel Pool Ventilation System," is changed regarding the demonstration of negative pressure during system operation.

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,

/S/

David H. Jaffe, Project Manager
PWR Project Directorate #8
Division of PWR Licensing-B

FOB
W. Regan
6/13/86

Enclosures:
1. Amendment No. 118 to DPR-53
2. Amendment No. 110 to DPR-69
3. Safety Evaluation

cc w/enclosure:
See next page

PBD#8	PBD#8	PBD#8	OELD
PMKreutzer	DJaffe	AThadani	
6/9/86	6/9/86	6/10/86	6/11/86

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Mr. J. A. Tiernan
Baltimore Gas & Electric Company

Calvert Cliffs Nuclear Power Plant

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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. 118
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 January 20, 1986 as supplemented by letter dated April 25, 1986, 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. 118, 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: June 17, 1986

ATTACHMENT TO LICENSE AMENDMENT NO. 118

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 7-5b	3/4 7-5b
3/4 7-25	3/4 7-25
3/4 7-26	3/4 7-26
3/4 9-15	3/4 9-15
B 3/4 7-2	B 3/4 7-2
B 3/4 7-5	B 3/4 7-5

AUXILIARY FEEDWATER SYSTEM

SURVEILLANCE REQUIREMENTS (Continued)

characteristics not required) and each auxiliary feedwater pump automatically starts upon receipt of each AFAS test signal, and

2. Verifying that the auxiliary feedwater system is capable of providing a minimum of 300 gpm nominal flow to each flow leg.*

*This surveillance may be performed on one flow leg at a time.

3/4.7.8 SNUBBERS

LIMITING CONDITION FOR OPERATION

3.7.8.1 All snubbers listed in Table 3.7-4 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.

---

* 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.

PLANT SYSTEMS

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.

SURVEILLANCE REQUIREMENTS (Continued)

d. At least once per 18 months by:

1. Verifying that the pressure drop across the combined HEPA filters and charcoal adsorber banks is < 4 inches Water Gauge while operating the ventilation system at a flow rate of 32,000 cfm ± 10%.

2. Verifying that each exhaust fan maintains the spent fuel storage pool area at a measurable negative pressure relative to the outside atmosphere during system operation.

e. After each complete or partial replacement of a HEPA filter bank by verifying that the HEPA filter banks remove ≥ 99% of the DOP when they are tested in-place in accordance with ANSI N510-1975 while operating the ventilation system at a flow rate of 32,000 cfm ± 10%.

f. After each complete or partial replacement of a charcoal adsorber bank by verifying that the charcoal adsorbers remove ≥ 99% of a halogenated hydrocarbon refrigerant test gas when they are tested in-place in accordance with ANSI N510-1975 while operating the ventilation system at a flow rate of 32,000 cfm ± 10%.

g. After maintenance affecting the air flow distribution by testing in-place and verifying that the air flow distribution is uniform within ± 20% of the average flow per unit when tested in accordance with the provisions of Section 9 of "Industrial Ventilation" and Section 8 of ANSI N510-1975.

SPENT FUEL CASK HANDLING CRANE

LIMITING CONDITION FOR OPERATION

3.9.13 Crane travel of the spent fuel shipping cask crane shall be restricted to prohibit a spent fuel shipping cask from travel over any area within one shipping cask length of any fuel assembly.

APPLICABILITY: With fuel assemblies in the storage pool.

ACTION:

With the requirements of the above specification not satisfied, place the crane load in a safe condition. The proviisons of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.13 Crane interlocks and physical stops which restrict a spent fuel shipping cask from passing over any area within one shipping cask length of any fuel assembly shall be demonstrated OPERABLE within 7 days prior to crane use and at least once per 7 days thereafter during crane operation.

3/4.7 PLANT SYSTEMS

BASES

3/4.7.1 TURBINE CYCLE

3/4.7.1.1 SAFETY VALVES

The OPERABILITY of the main steam line code safety valves ensures that the secondary system pressure will be limited to within 110% of its design pressure of 1000 psig during the most severe anticipated system operational transient. The total relieving capacity for all valves on all of the steam lines is 12.18×10^6 lbs/hr at 100% RATED THERMAL POWER. The maximum relieving capacity is associated with a turbine trip from 100% RATED THERMAL POWER coincident with an assumed loss of condenser heat sink (i.e., no steam bypass to the condenser). The main steam line code safety valves are tested and maintained in accordance with the requirements of Section XI of the ASME Boiler and Pressure Vessel Code. The as-left lift settings will be no less than 985 psig to ensure that the lift setpoints will remain within specification during the cycle.

In MODE 3, two main steam safety valves are required OPERABLE per steam generator. These valves will provide adequate relieving capacity for removal of both decay heat and reactor coolant pump heat from the reactor coolant system via either of the two steam generators. This requirement is provided to facilitate the post-overhaul setting and OPERABILITY testing of the safety valves which can only be conducted when the RCS is at or above 500°F. It allows entry into MODE 3 with a minimum number of main steam safety valves OPERABLE so that the set pressure for the remaining valves can be adjusted in the plant. This is the most accurate means for adjusting safety valve set pressures since the valves will be in thermal equilibrium with the operating environment.

STARTUP and/or POWER OPERATION is allowable with safety valves inoperable within the limitations of the ACTION requirements on the basis of the reduction in secondary system steam flow and THERMAL POWER required by the reduced reactor trip settings of the Power Level-High channels. The reactor trip setpoint reductions are derived on the following bases:

For two loop operation

$$SP = \frac{(X) - (Y)(V)}{X} \times 106.5$$

For single loop operation (two reactor coolant pumps operating in the same loop)

$$SP = \frac{(X) - (Y)(U)}{X} \times 46.8$$

where:

SP = reduced reactor trip setpoint in percent of RATED THERMAL POWER

V = maximum number of inoperable safety valves per steam line

U = maximum number of inoperable safety valves per operating steam line

106.5 = Power Level - High Trip Setpoint for two loop operation

46.8 = Power Level - High Trip Setpoint for single loop operation with two reactor coolant pumps operating in the same loop

X = Total relieving capacity of all safety valves per steam line in lbs/hour

Y = Maximum relieving capacity of any one safety valve in lbs/hour

3/4.7.1.2 AUXILIARY FEEDWATER SYSTEM

The OPERABILITY of the auxiliary feedwater system ensures that the Reactor Coolant System can be cooled down to less than 300°F from normal operating conditions in the event of a total loss of offsite power. A delivered flow of 300 gpm is sufficient to ensure that adequate feedwater flow is available to remove decay heat and reduce the Reactor Coolant System temperature to less than 300°F when the shutdown cooling system may be placed into operation.

Flow control valves were installed in the system in order to allow automatic flow initiation to a value selected by the Operator. Maximum flow to the steam generators from the motor driven AFW pump powered from the diesel is 300 gpm when feeding both generators (i.e., 150 gpm per leg maximum flow). The flow control valves installed in each leg supplied from the motor driven AFW pump shall be set at a flow setpoint not to exceed 150 gpm per leg. If the flow is only being directed to one steam generator, it is acceptable to deliver a maximum of 330 gpm because the flow error associated with the non-used loop is eliminated. These motor driven AFW pump capacity limits are imposed to prevent exceeding the emergency diesel generator load limit. If diesel generator loading is not a limiting concern, the delivered flow from the motor driven AFW pump may be increased up to a maximum of 575 gpm (motor HP limit vice diesel loading limit). These upper flow limits do not apply to the steam driven pumps.

In the spectrum of events analyzed in which automatic initiation of auxiliary feedwater occurs, the following flow conditions are allowed with an operator action time of 10 minutes.

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 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.

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. 110
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 January 20, 1986 as supplemented by letter dated April 25, 1986, 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. Technical Specifications

      The Technical Specifications contained in Appendix A, as revised through Amendment No. 110, 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: June 17, 1986

ATTACHMENT TO LICENSE AMENDMENT NO. 110

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 7-5a	3/4 7-5a
3/4 7-25	3/4 7-25
3/4 7-26	3/4 7-26
3/4 9-15	3/4 9-15
B 3/4 7-2	B 3/4 7-2
B 3/4 7-5	B 3/4 7-5

AUXILIARY FEEDWATER SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.1.2 Two auxiliary feedwater trains consisting of one steam driven and one motor driven pump and associated flow paths capable of automatically initiating flow shall be OPERABLE. (An OPERABLE steam driven train shall consist of one pump aligned for automatic flow initiation and one pump aligned in standby.)*

APPLICABILITY: MODES 1, 2 and 3.

ACTION:

a. With any single pump inoperable, perform the following:

1. With No. 23 motor-driven pump inoperable:

(a) Align the standby steam-driven pump to automatic initiating status within 72 hours or be in HOT SHUTDOWN within the next 12 hours, and

(b) Restore No. 23 motor-driven pump to OPERABLE status within the next 7 days or be in HOT SHUTDOWN within the next 12 hours.

2. With one steam-driven pump inoperable:

(a) Align the OPERABLE steam driven pump to automatic initiating status within 72 hours or be in HOT SHUTDOWN within the next 12 hours, and

(b) Restore the inoperable steam driven pump to standby status (or automatic initiating status if the other steam driven pump is to be placed in standby) within the next 7 days or be in HOT SHUTDOWN within the next 12 hours.

b. With any two pumps inoperable:

1. Verify that the remaining pump is aligned to automatic initiating status within one hour, and

2. Verify within one hour that No. 13 motor driven pump is OPERABLE and valve 1-CV-4550 has been exercised within the last 30 days, and

3. Restore a second pump to automatic initiating status within 72 hours or be in HOT SHUTDOWN within the next 12 hours.

*A standby pump shall be available for operation but aligned so that automatic flow initiation is defeated upon AFAS actuation.

LIMITING CONDITION FOR OPERATION (Continued)

c. Whenever a sybsystem(s) (a subsystem consisting of one pump, piping, valves and controls in the direct flow path) required for operability is inoperable for the performance of periodic testing (e.g., manual discharge valve closed for pump Total Dynamic Head Test or Logic Testing) a dedicated operator(s) will be stationed at the local station(s) with direct communication to the Control Room. Upon completion of any testing, the subsystem(s) required for operability will be returned to its proper status and verified in its proper status by an independent operator check.

d. The requirements of Specification 3.0.4 are not applicable whenever one motor and one steam-driven pump (or two steam-driven pumps) are aligned for automatic flow initiation.

SURVEILLANCE REQUIREMENTS

4.7.1.2 Each auxiliary feedwater flowpath shall be demonstrated OPERABLE:

a. At least once per 31 days by:

1. Verifying that each steam driven pump develops a Total Dynamic Head of $\geq$ 2800 ft. on recirculation flow. (If verification must be demonstrated during startup, surveillance testing shall be performed upon achieving an RCS temperature $\geq$ 300°F and prior to entering MODE 1).

2. Verifying that the motor drivem pump develops a Total Dynamic Head of $\geq$ 3100 ft. on recirculation flow.

3. Cycling each testable, remote operated valve that is not in its operating position through at least one complete cycle.

4. Verifying that each valve (manual, power operated or automatic) in the direct flow path is in its correct position.

b. Before entering MODE 3 after a COLD SHUTDOWN of at least 14 days by completing a flow test that verifies the flow path from the condensate storage tank to the steam generators.

c. At least once per 18 months by:

1. Verifying that each automatic valve in the flow path actuates to its correct position (verification of flow-modulating characteristics not required) and each auxiliary feedwater pump automatically starts upon receipt of each AFAS test signal, <u>and</u>

2. Verifying that the auxiliary feedwater system is capable of providing a minimum of 300 gpm nominal flow to each flow leg.*

---

*This surveillance may be performed on one flow leg at a time.

PLANT SYSTEMS

3/4.7.8 SNUBBERS

LIMITING CONDITION FOR OPERATION

3.7.8.1 All snubbers listed in Table 3.7-4 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.

* 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.

SURVEILLANCE REQUIREMENTS (Continued)

d. At least once per 18 months by:

   1. Verifying that the pressure drop across the combined HEPA filters and charcoal adsorber banks is < 4 inches Water Gauge while operating the ventilation system at a flow rate of 32,000 cfm ± 10%.

   2. Verifying that each exhaust fan maintains the spent fuel storage pool area at a measurable negative pressure relative to the outside atmosphere during system operation.

e. After each complete or partial replacement of a HEPA filter bank by verifying that the HEPA filter banks remove ≥ 99% of the DOP when they are tested in-place in accordance with ANSI N510-1975 while operating the ventilation system at a flow rate of 32,000 cfm ± 10%.

f. After each complete or partial replacement of a charcoal adsorber bank by verifying that the charcoal adsorbers remove ≥ 99% of a halogenated hydrocarbon refrigerant test gas when they are tested in-place in accordance with ANSI N510-1975 while operating the ventilation system at a flow rate of 32,000 cfm ± 10%.

g. After maintenance affecting the air flow distribution by testing in-place and verifying that the air flow distribution is uniform within ± 20% of the average flow per unit when tested in accordance with the provisions of Section 9 of "Industrial Ventilation" and Section 8 of ANSI N510-1975.

SPENT FUEL CASK HANDLING CRANE

LIMITING CONDITION FOR OPERATION

3.9.13 Crane travel of the spent fuel shipping cask crane shall be restricted to prohibit a spent fuel shipping cask from travel over any area within one shipping cask length of any fuel assembly.

APPLICABILITY: With fuel assemblies in the storage pool.

ACTION:

With the requirements of the above specification not satisfied, place the crane load in a safe condition. The proviisons of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.13 Crane interlocks and physical stops which restrict a spent fuel shipping cask from passing over any area within one shipping cask length of any fuel assembly shall be demonstrated OPERABLE within 7 days prior to crane use and at least once per 7 days thereafter during crane operation.

3/4.7 PLANT SYSTEMS

BASES

3/4.7.1 TURBINE CYCLE

3/4.7.1.1 SAFETY VALVES

The OPERABILITY of the main steam line code safety valves ensures that the secondary system pressure will be limited to within 110% of its design pressure of 1000 psig during the most severe anticipated system operational transient. The total relieving capacity for all valves on all of the steam lines is 12.18×10^6 lbs/hr at 100% RATED THERMAL POWER. The maximum relieving capacity is associated with a turbine trip from 100% RATED THERMAL POWER coincident with an assumed loss of condenser heat sink (i.e., no steam bypass to the condenser). The main steam line code safety valves are tested and maintained in accordance with the requirements of Section XI of the ASME Boiler and Pressure Code. The as-left lift settings will be no less than 985 psig to ensure that the lift setpoints will remain within specification during the cycle.

In MODE 3, two main steam safety valves are required OPERABLE per steam generator. These valves will provide adequate relieving capacity for removal of both decay heat and reactor coolant pump heat from the reactor coolant system via either of the two steam generators. This requirement is provided to facilitate the post-overhaul setting and operability testing of the safety valves which can only be conducted when the RCS is at or above 500°F. It allows entry into MODE 3 with a minimum number of main steam safety valves OPERABLE so that the set pressure for the remaining valves can be adjusted in the plant. This is the most accurate means for adjusting safety valve set pressures since the valves will be in thermal equilibrium with the operating environment.

STARTUP and/or POWER OPERATION is allowable with safety valves inoperable within the limitations of the ACTION requirements on the basis of the reduction in secondary system steam flow and THERMAL POWER required by the reduced reactor trip settings of the Power Level-High channels. The reactor trip setpoint reductions are derived on the following bases:

For two loop operation

$$SP = \frac{(X) - (Y)(V)}{X} \times 106.5$$

For single loop operation (two reactor coolant pumps operating in the same loop)

$$SP = \frac{(X) - (Y)(U)}{X} \times 46.8$$

where:

SP = reduced reactor trip setpoint in percent of RATED THERMAL POWER

V = maximum number of inoperable safety valves per steam line

U = maximum number of inoperable safety valves per operating steam line

106.5 = Power Level - High Trip Setpoint for two loop operation

46.8 = Power Level - High Trip Setpoint for single loop operation with two reactor coolant pumps operating in the same loop

X = Total relieving capacity of all safety valves per steam line in lbs/hour

Y = Maximum relieving capacity of any one safety valve in lbs/hour

3/4.7.1.2 AUXILIARY FEEDWATER SYSTEM

The OPERABILITY of the auxiliary feedwater system ensures that the Reactor Coolant System can be cooled down to less than 300°F from normal operating conditions in the event of a total loss of offsite power. A delivered flow of 300 gpm is sufficient to ensure that adequate feedwater flow is available to remove decay heat and reduce the Reactor Coolant System temperature to less than 300°F when the shutdown cooling system may be placed into operation.

Flow control valves were installed in the system in order to allow automatic flow initiation to a value selected by the Operator. Maximum flow to the steam generators from the motor driven AFW pump powered from the diesel is 300 gpm when feeding both generators (i.e., 150 gpm per leg maximum flow). The flow control valves installed in each leg supplied from the motor driven AFW pump shall be set at a flow setpoint not to exceed 150 gpm per leg. If the flow is only being directed to one steam generator, it is acceptable to deliver a maximum of 330 gpm because the flow error associated with the non-used loop is eliminated. These motor driven AFW pump capacity limits are imposed to prevent exceeding the emergency diesel generator load limit. If diesel generator loading is not a limiting concern, the delivered flow from the motor driven AFW pump may be increased up to a maximum of 575 gpm (motor HP limit vice diesel loading limit). These upper flow limits do not apply to the steam driven pumps.

In the spectrum of events analyzed in which automatic initiation of auxiliary feedwater occurs, the following flow conditions are allowed with an operator action time of 10 minutes.

(1)	Loss of Feedwater	0 gpm Auxiliary Feedwater Flow
(2)	Feedline Break	0 GPM Auxiliary Feedwater Flow

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 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.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NOS.118 AND 110

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 application for license amendments dated January 20, 1986, as supplemented by letter dated April 25, 1986, Baltimore Gas and Electric Company (BG&E) requested changes to the Technical Specifications (TS) for Calvert Cliffs Units 1 and 2. The proposed amendments would change the Unit 1 and Unit 2 TS as follows: (1) TS 4.7.1.2c.2, "Auxiliary Feedwater System," would be changed to reflect a revised flow test requirement. In addition, a typographical error would be corrected (Unit 2 only); (2) TS 4.7.8.1, "Snubbers," would be changed to subdivide the snubber surveillance population based upon snubber type; and (3) TS 4.9.12d.2, "Spent Fuel Pool Ventilation System," would be changed regarding the demonstration of negative pressure during system operation.

Discussion and Evaluation

BG&E has proposed a change to TS 4.7.1.1.2c.2 and the associated Bases that would increase the 18 month demonstrated auxiliary feedwater (AFW) flow from 200 gpm to 300 gpm. A footnote would also be added which would clarify the flow requirement to indicate that the "...surveillance may be performed on one flow leg at a time." A flow leg is a flow path leading from the discharge of an AFW pump to one of two steam generators. The existing TS 4.7.1.2c.2 requires demonstration of "...a minimum of 200 gpm nominal flow to each flow leg."

The proposed change to TS 4.7.1.2c.2 and the associated Bases was necessitated by modifications to the motor driven AFW pump recirculation. The pump recirculation provides a flow path to supply cooling water, from pump discharge to the pump intake, to prevent pump damage during the period from initial pump operation to onset of pump flow. The modification to the recirculation path for the motor driven AFW pump allows the automatic recirculation valve to be overriden. In this condition the valve will be in permanent recirculation. The valve was originally installed to decrease the loading on the diesels by eliminating the continuous recirculation from the pump. As the AFW system was tested, it became apparent that with automatic auxiliary feedwater actuation at low steam generator pressure, the recirculation valve and flow control valves were causing a flow instability problem.

Corrective actions for the AFW flow instability problem are relatively straightforward; however, their impact on emergency diesel generator loading has had to be closely studied. Placing the automatic recirculation valve in permanent recirculation without restricting delivered flow to the steam generators would cause the pump to require additional power. This increased load would exceed the maximum diesel generator load that has been dedicated to AFW. Therefore, calculations were performed to reanalyze the minimum required long term AFW flowrate. The result of these calculations showed that 300 gpm delivered to a single steam generator, or distributed between both steam generators, is sufficient to remove decay heat from the primary system from full-power post trip conditions to a primary system temperature of 300°F, after which shutdown cooling can be used to continue the cool-down process. The above assumes that off-site power has been lost. Previous calculations had indicated that a minimum AFW flow of 400 gpm was required. The 300 gpm value was attained by crediting initiation of charging flow (one pump) at 60 minutes, reanalyzing the reactor coolant system (RCS) heat capacity, and applying ANS-5 1971 for decay heat loads (as permitted by 10 CFR Part 50, Appendix K). With this AFW delivered flow, all decay heat can be removed and the RCS cooled down to less than 300°F from normal operating conditions in the event of a total loss of offsite power. This cooldown is accomplished within the 6-hour criteria of the Calvert Cliffs FSAR Section 10.3.2.

The output of the AFW system is not limited to the capacity of the motor-driven AFW pump. In addition to the 300 gpm provided by the motor-driven AFW pump, two steam turbine driven AFW pumps are available, each with a design flow of 700 gpm at 2490 feet total dynamic head (TDH). In addition, should a motor-driven AFW pump become inoperable, the motor-driven AFW pump from the unaffected unit can be cross-connected to provide additional flow capability. However, the demonstration that the motor-driven AFW pump, alone, is sufficient to cool down the RCS to 300°F, maintains the motor-driven pump as a 100% capacity component and represents the principal motivation of the licensee for recalculation of the required AFW flow rate.

With the automatic recirculation valve in permanent recirculation, adequate AFW flow is provided to meet the FSAR and TS heat removal bases. Additionally, by imposing a maximum flow limit on the motor-driven AFW pump, the design emergency diesel generator loading is not exceeded.

Finally, calculations performed indicate that 300 gpm supplied from the motor-driven AFW pump is sufficient to ensure that adequate AFW flow is available to remove decay heat and reduce the RCS temperature to less than 300°F from normal operating conditions in the event of a total loss of offsite power. The Calvert Cliffs FSAR references Loss of Feedwater (LOFW) and Feedline Break (FLB) as undercooling scenarios. (FLB is not a design basis event for Calvert Cliffs, but it was analyzed for peak RCS pressure and 10 CFR Part 100 boundary dose calculations in association with the addition of automatic initiation of AFW). Undercooling events are of interest since they may be sensitive to AFW flow. For the LOFW event, the FSAR assumes that the Operator initiates AFW flow at 600 seconds with no minimum flow specified. The Operator, as stated in the bases, is assumed to be able to increase or decrease AFW flow to that required by existing plant conditions. Feedline Break is analyzed for peak RCS pressure and credits no AFW flow until well after the peak pressure has occurred. Based upon the above, the proposed changes to TS 4.7.1.2c.2 and the associated Bases is acceptable.

BG&E has proposed a change to TS 4.7.1.2c (Unit 2, only) that corrects a typographical error. The TS section designated as 4.7.1.2.c.a should be 4.7.1.2.c.1.

The proposed change to TS 4.7.1.2c does correct an error in the TS and thus the proposed change is acceptable.

BG&E has proposed a change to TS 4.7.8.1 and the associated Bases in order to differentiate between "types" of snubbers for the purpose of surveillance. At the present time, TS 4.7.8.1. requires a representative sample of 10% of the snubbers to undergo surveillance on a sliding scale, depending on snubber performance, ranging from 18 months (no failures) to 31 days (eight or more failures). Moreover, should a snubber fail the "functional" test, an additional 5% of the snubbers must be functionally tested. BG&E proposes to define snubber "type" as follows: "As used in this specification, type of snubber shall mean snubbers of the same design and manufacturer, irrespective of capacity." The definition of "type" would also be incorporated in the functional test requirements of TS 4.7.8.1c such that snubber failures would result in additional tests for only a particular "type" of snubber rather than the general snubber population.

BG&E currently utilizes two types of hydraulic snubbers, both produced by the same manufacturer (Grinnell). The two types operate similarly in principle, but have different design features (irrespective of capacity related dimension differences). Small bore snubbers installed on various piping systems at Calvert Cliffs have bore sizes that range from 1-1/2" to 6". All small bore snubbers have the same design valve block and have only one valve block per snubber. The large bore snubbers, installed on the steam generators at Calvert Cliffs, all have 10" bores. The valve blocks on these snubbers are different in design from those on the small bore snubbers. Additionally, there are two valve blocks per snubber.

The differences between these two types of snubbers is apparent upon a review of functional testing load and acceptance criteria. Small bore snubbers are designed for loads up to 72,000 pounds force. The Locking Velocity (LV) and Bleed Rate (BR) acceptance criteria are as follows:

<u>Inches Per Minute (IPM)</u>

LV: 1-40
BR: .25-25 (Adjusted for room temperature)

The large bore snubbers are designed for loads up to 300,000 pounds force. Their acceptance criteria are much more restrictive:

Inches Per Minute (IPM)

LV: 1.25-1.75
BR: 0.0625-0.1875

Based upon the above, a functional failure in the small bore snubber population is not significant with regard to the large bore snubber population and vice versa.

Although all snubbers at Calvert Cliffs are currently produced by the same manufacturer, Surveillance Requirement 4.7.8.1 was expanded using standard industry phraseology to define "type" as "of the same design and manufacturer." Inclusion of the word "manufacturer" precludes the necessity of a future TS change should BG&E replace certain snubbers with a different make. The difference between small and large bore hydraulic snubbers is explicitly referenced in the proposed TS Bases to prevent interpretive difficulties.

Differentiation by snubber "type" with regard to snubber surveillance has been an NRC position for some time and is reflected in the NRC's Combustion Engineering Standard Technical Specifications.

Based upon the above, we conclude that the proposed change to TS 4.7.8.1 and associated Bases are acceptable.

BG&E has proposed a change to TS 4.9.12.d.2, which requires verification that each exhaust fan in the Spent Fuel Pool Ventilation System maintains a negative pressure in the spent fuel storage pool area. Currently, TS 4.9.12.d.2 requires that a "negative pressure greater than or equal to 1/8 inches Water Gauge" be maintained by each exhaust fan. The proposed change would require that a "measurable negative" pressure be maintained by each exhaust fan.

A minimum allowable "measurable negative" pressure will be established for the surveillance and will be controlled administratively. The differential pressure established will be verified, by smoke test of area access doors and hatches, to be negative enough to assure that, where leakage occurs, the air flow path is into the spent fuel pool area.

The reactor containment under refueling conditions and the spent fuel pool area are similar in that, in both cases, the degree of containment is based upon a fuel handling accident (i.e. the dropping of a spent fuel assembly within the refueling/storage pool.) The Bases for TS 3/4.9.4, "Containment Penetrations", states that "The OPERABILITY and closure restrictions are sufficient to restrict radioactive material release from a fuel element rupture based upon the lack of containment pressurization potential while in the REFUELING MODE." Similarly, a fuel handling accident in the spent fuel pool area would not result in a pressurization potential. A "measurable negative pressure" maintained by the spent fuel pool venilitation system is therefore sufficient to assure that any material that is released will be treated by the filters in the ventilation system.

The design basis for the Spent Fuel Pool Ventilation System is to ensure that all radioactive material released from an irradiated fuel assembly will be filtered through the HEPA filters and charcoal adsorbers prior to discharge to the atmosphere. The system is assumed to perform this function for a Fuel Handling Incident as mentioned in the FSAR, Section 14.18. A "measurable negative" pressure in the spent fuel storage pool area will continue to assure that air flow is into this area from outside. Therefore, the amount of radioactive material released during a Fuel Handling Incident would not be significantly increased. Based upon the above, the proposed change to TS 4.9.12.d.2 is acceptable.

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 or changes in surveillance requirements. 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 considerations 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). Pursuant to 10 CFR §51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendments.

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.

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