December 4, 1985

Docket Nos. 50-317
and 50-318

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Mr. A. E. Lundvall, Jr.
Vice President - Supply
Baltimore Gas & Electric Company
P. O. Box 1475
Baltimore, Maryland 21203

Dear Mr. Lundvall:

The Commission has issued the enclosed Amendment Nos. 108 and 91 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 response to your application dated June 28, 1985.

The amendments would chang the Unit 1 and Unit 2 Technical Specifications (TS) to: (1) reflect a clarification of requirements associated with the containment purge isolation valves in TS Table 3.3-3, "Engineered Safety Features Actuation System Instrumentation," and TS Table 3.6-1, "Containment Isolation Valves," (2) modify TS 3.9.4, "Containment Penetration," to allow the use of an alternate closure for the emergency personnel escape lock, (3) delete TS 6.13, "Environmental Qualifications," and (4) correct identified spelling errors and changes in terminology. Additional changes to the TS that were requested in your June 28, 1985 application will be addressed in subsequent correspondence.

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

Sincerely,

/S/

David H. Jaffe, Project Manager
Operating Reactors Branch #3
Division of Licensing

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

cc w/enclosure:
See next page

PWRPD#8	PWRPD#8	PWRPD#8	OELD	~~D:PWRPD-B~~
PMKreutzer	DJaffe:dd	AThadani		~~FMiraglia~~
11/27/85	11/27/85	11/27/85	12/3/85	~~7/85~~

Mr. A. E. Lundvall, Jr.
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. 108
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 June 28, 1985 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. 108 , 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 4, 1985

ATTACHMENT TO LICENSE AMENDMENT NO. 108

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
XVI	XVI
3/4 1-21	3/4 1-21
3/4 1-25	3/4 1-25
3/4 3-13	3/4 3-13
3/4 3-16	3/4 3-16
3/4 4-8	3/4 4-8
3/4 6-25	3/4 6-25
3/4 7-72	3/4 7-72
3/4 9-4	3/4 9-4
3/4 9-11	3/4 9-11
3/4 10-5	3/4 10-5
B 3/4 1-3	B 3/4 1-3
B 3/4 9-3	B 3/4 9-3
6-14	6-14
6-21	6-21
6-22	6-22

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ADMINISTRATIVE CONTROLS

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ADMINISTRATIVE CONTROLS

LIMITING CONDITION FOR OPERATION

3.1.3.3 At least two of the following three CEA position indicator channels shall be OPERABLE for each shutdown and regulating CEA:

a. CEA voltage divider reed switch position indicator channel, capable of determining the absolute CEA position within ± 1.75 inches;

b. CEA "Full Out" or "Full In" reed switch position indicator channel, only if the CEA is fully withdrawn or fully inserted, as verified by actuation of the applicable position indicator; and

c. CEA pulse counting position indicator channel.

APPLICABILITY: MODES 1 and 2.

ACTION:

a. With a maximum of one CEA per group having its voltage divider reed switch position indicator channel or its pulse counting position indicator channel inoperable and the CEA(s) with the inoperable position indicator channel partially inserted, either:

1. Within 6 hours

a) Restore the inoperable position indicator channel to OPERABLE status, or

b) Be in at least HOT STANDBY, or

c) Reduce THERMAL POWER to ≤ 70% of the maximum allowable THERMAL POWER level for the existing Reactor Coolant Pump combination; if negative reactivity insertion is required to reduce THERMAL POWER, boration shall be used. Operation at or below this reduced THERMAL POWER level may continue provided that within the next 4 hours either:

1) The CEA group(s) with the inoperable position indicator is fully withdrawn while maintaining the withdrawal sequence required by Specification 3.1.3.6 and when this CEA group reaches its fully withdrawn position, the "Full Out" limit of the CEA with the inoperable position indicator is actuated and verifies this CEA to be fully withdrawn. Subsequent to fully withdrawing this CEA group(s), the THERMAL POWER level may be returned to a level consistent with all other applicable specifications and operation may continue per Specification 3.1.3.3 above; or

2) The CEA group(s) with the inoperable position indicator is fully inserted, and subsequently maintained fully inserted, while maintaining the withdrawal sequence and THERMAL POWER level required by Specification 3.1.3.6 and when this CEA group reaches its fully

LIMITING CONDITION FOR OPERATION

inserted position, the "Full In" limit of the CEA with the inoperable indicator is actuated and verifies this CEA to be fully inserted. Subsequent operation shall be within the limits of Specification 3.1.3.6, and may continue per Specification 3.1.3.3 above.

2. or, if the failure existed before entry into MODE 2 or occurs prior to an "all CEAs out" configuration, the CEA group(s) with inoperable position indicator channel must be moved to the "Full Out" position and verified to be fully withdrawn via a "Full Out" indicator. These actions must be completed within 10 hours of entry into MODE 2 and prior to exceeding 70% of the maximum allowable THERMAL POWER level for the existing Reactor Coolant Pump combination. The provisions of Specification 3.0.4 are not applicable. Once these actions are completed, operation may continue per Specification 3.1.3.3 above.

b. With more than one CEA per group having its CEA pulse counting position indicator channel and either (1) the "Full Out" or "Full In" position indicator, or (2) the voltage divider position indicator channel inoperable, operation in MODES 1 and 2 may continue for up to 24 hours provided that for the affected CEAs, either:

1. The CEA voltage divider reed switch position indicator channels are OPERABLE, or

2. The CEA "Full Out" or "Full In" reed switch position indicator channels are OPERABLE, with the CEA fully withdrawn or fully inserted as verified by actuation of the applicable position indicator.

SURVEILLANCE REQUIREMENTS

4.1.3.3.1 Each required CEA position indication channel shall be determined to be OPERABLE by determining CEA positions as follows at least once per 12 hours, by:

a. Verifying the CEA pulse counting position indicator channels and the CEA voltage divider reed switch position indicator channels agree within 4.5 inches, or

b. Verifying the CEA pulse counting position indicator channels and the CEA "Full Out" or "Full In" reed switch position indicator channels agree within 4.5 inches, or

c. Verifying the CEA voltage divider reed switch position indicator channels and the CEA "Full Out" or "Full In" reed switch position indicator channels agree within 4.5 inches.

4.1.3.3.2 During time intervals when the deviation circuit is inoperable, the above verification of required CEA position indicator channels shall be made at least once per 4 hours.

REGULATING CEA INSERTION LIMITS

LIMITING CONDITION FOR OPERATION

3.1.3.6 The regulation CEA groups shall be limited to the withdrawal sequence and to the insertion limits shown on Figure 3.1-2 (regulating CEAs are considered to be fully withdrawn in accordance with Figure 3.1-2 when withdrawn to at least 129.0 inches) with CEA insertion between the Long Term Steady State Insertion Limits and the Transient Insertion Limits restricted to:

a. ≤ 4 hours per 24 hour interval,

b. ≤ 5 Effective Full Power Days per 30 Effective Full Power Day interval, and

c. ≤ 14 Effective Full Power Days per calendar year.

APPLICABILITY: MODES 1* and 2*#.

ACTION:

a. With the regulating CEA groups inserted beyond the Transient Insertion Limits, except for surveillance testing pursuant to Specification 4.1.3.1.2, within two hours either:

   1. Restore the regulating CEA groups to within the limits, or

   2. Reduce THERMAL POWER to less than or equal to that fraction of RATED THERMAL POWER which is allowed by the CEA group position using Figure 3.1-2.

b. With the regulating CEA groups inserted between the Long Term Steady State Insertion Limits and the Transient Insertion Limits for intervals > 4 hours per 24 hour interval, except during operations pursuant to the provisions of ACTION items c. and e. of Specification 3.1.3.1, operation may proceed provided either:

   1. The Short Term Steady State Insertion Limits of Figure 3.1-2 are not exceeded, or

   2. Any subsequent increase in THERMAL POWER is restricted to ≤ 5% of RATED THERMAL POWER per hour.

---

* See Special Test Exceptions 3.10.2 and 3.10.4.

With $K_{eff} \geq 1.0$.

LIMITING CONDITION FOR OPERATION (Continued)

c. With the regulating CEA groups inserted between the Long Term Steady State Insertion Limits and the Transient Insertion Limits for intervals > 5 EFPD per 30 EFPD interval or > 14 EFPD per calendar year, except during operations pursuant to the provisions of ACTION items c. and e. of Specification 3.1.3.1, either:

1. Restore the regulating groups to within the Long Term Steady State Insertion Limits within two hours, or

2. Be in at least HOT STANDBY within 6 hours.

SURVEILLANCE REQUIREMENTS

4.1.3.6 The position of each regulating CEA group shall be determined to be within the Transient Insertion Limits at least once per 12 hours except during time intervals when the PDIL Alarm Circuit is inoperable, then verify the individual CEA positions at least once per 4 hours. The accumulated times during which the regulating CEA groups are inserted beyond the Steady State Insertion Limits but within the Transient Insertion Limits shall be determined at least once per 24 hours.

TABLE 3.3-3 (Continued)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
6. CONTAINMENT PURGE VALVES ISOLATION ##					
a. Manual (Purge Valve Control Switches)	2/Penetration	1/Penetration	2/Penetration	6*	8
b. Containment Radiation - High Area Monitor	4	2	3	6*	8
7. LOSS OF POWER					
a. 4.16 kv Emergency Bus Undervoltage (Loss of Voltage)	4/Bus	2/Bus	3/Bus	1, 2, 3	7*
b. 4.16 kv Emergency Bus Undervoltage (Degraded Voltage)	4/Bus	2/Bus	3/Bus	1, 2, 3	7*

Containment purge valve isolation is also initiated by SIAS (functional units 1.a, 1.b, and 1.c).

* Must be OPERABLE only in MODE 6 when the valves are required OPERABLE and they are open.

TABLE 3.3-3 (Continued)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
8. CVCS ISOLATION					
a. Manual (CVCS Isolation Valve Control Switches)	1/Valve	1/Valve	1/Valve	1, 2, 3, 4	6
b. West Penetration Room/Letdown Heat Exchanger Room Pressure - High	4	2	3	1, 2, 3, 4	7*
9. AUXILIARY FEEDWATER ACTUATION SYSTEM (AFAS)					
a. Manual (Trip Buttons)	2 sets of 2 per S/G	1 set of 2 per S/G	2 sets of 2 per S/G	1, 2, 3	6
b. Steam Generator Level - Low	4/SG	2/SG	3/SG	1, 2, 3	7
c. Steam Generator ΔP High	4/SG	2/SG	3/SG	1, 2, 3	7

TABLE 3.3-3 (Continued)

TABLE NOTATION

(a) Trip function may be bypassed in this MODE when pressurizer pressure is < 1800 psia; bypass shall be automatically removed when pressurizer pressure is ≥ 1800 psia.

(c) Trip function may be bypassed in this MODE below 785 psia; bypass shall be automatically removed at or above 785 psia.

* The provisions of Specification 3.0.4 are not applicable.

ACTION STATEMENTS

ACTION 6 - With the number of OPERABLE channels one less than the Total Number of Channels, restore the inoperable channel to OPERABLE status within 48 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

ACTION 7 - With the number of OPERABLE channels one less than the Total Number of Channels, operation may proceed provided the following conditions are satisfied:

a. The inoperable channel is placed in either the bypassed or tripped condition within 1 hour. For the purposes of testing and maintenance, the inoperable channel may be bypassed for up to 48 hours from time of initial loss of OPERABILITY; however, the inoperable channel shall then be either restored to OPERABLE status or placed in the tripped condition.

b. Within one hour, all functional units receiving an input from the inoperable channel are also placed in the same condition (either bypassed or tripped, as applicable) as that required by a. above for the inoperable channel.

c. The Minimum Channels OPERABLE requirement is met; however, one additional channel may be bypassed for up to 48 hours while performing tests and maintenance on that channel provided the other inoperable channel is placed in the tripped condition.

TABLE 3.3-3 (Continued)

ACTION 8 - With less than the Minimum Channels OPERABLE, operation may continue provided the containment purge valves are maintained closed.

ACTION 11 - With the number of OPERABLE Channels one less than the Total Number of Channels, operation may proceed provided the inoperable channel is placed in the bypassed condition and the Minimum Channels OPERABLE requirement is demonstrated within 1 hour; one additional channel may be bypassed for up to 2 hours for surveillance testing per Specification 4.3.2.1.

2. Tubes in those areas where experience has indicated potential problems.

c. The second and third inservice inspections may be less than a full tube inspection by concentrating (selecting at least 50% of the tubes to be inspected) the inspection on those areas of the tube sheet array and on those portions of the tubes where tubes with imperfections were previously found.

The results of each sample inspection shall be classified into one of the following three categories:

Category	Inspection Results
C-1	Less than 5% of the total tubes inspected are degraded tubes and none of the inspected tubes are defective.
C-2	One or more tubes, but not more than 1% of the total tubes inspected are defective, or between 5% and 10% of the total tubes inspected are degraded tubes.
C-3	More than 10% of the total tubes inspected are degraded tubes or more than 1% of the inspected tubes are defective.

Note: In all inspections, previously degraded tubes must exhibit significant (>10%) further wall penetrations to be included in the above percentage calculations.

4.4.5.3 Inspection Frequencies - The above required inservice inspections of steam generator tubes shall be performed at the following frequencies:

a. The first inservice inspection shall be performed after 6 Effective Full Power Months but within 24 calendar months of initial criticality. Subsequent inservice inspections shall be performed at intervals of not less than 12 nor more than 24 calendar months after the previous inspection. If two consecutive inspections following service under AVT conditions, not including the preservice inspection, result in all inspection results falling into the C-1 category or if two consecutive inspections demonstrate that previously observed degradation has not continued and no additional degradation has occurred, the inspection interval may be extended to a maximum of once per 40 months.

SURVEILLANCE REQUIREMENTS (Continued)

b. If the inservice inspection of a steam generator conducted in accordance with Table 4.4-2 requires a third sample inspection whose results fall in Category C-3, the inspection frequency shall be reduced to at least once per 20 months. The reduction in inspection frequency shall apply until a subsequent inspection demonstrates that a third sample inspection is not required.

c. Additional, unscheduled inservice inspections shall be performed on each steam generator in accordance with the first sample inspection specified in Table 4.4-2 during the shutdown subsequent to any of the following conditions:

1. Primary-to-secondary tube leaks (not including leaks originating from tube-to-tube sheet welds) in excess of the limits of Specification 3.4.6.2,

2. A seismic occurrence greater than the Operating Basis Earthquake,

3. A loss-of-coolant accident requiring actuation of the engineered safeguards, or

4. A main steam line or feedwater line break.

4.4.5.4 Acceptance Criteria

a. As used in this Specification:

1. Imperfection means an exception to the dimensions, finish or contour of a tube from that required by fabrication drawings or specifications. Eddy-current testing indications below 20% of the nominal tube wall thickness, if detectable, may be considered as imperfections.

2. Degradation means a service-induced cracking, wastage, wear or general corrosion occurring on either inside or outside of a tube.

3. Degraded Tube means a tube containing imperfections $\geq$20% of the nominal wall thickness caused by degradation.

4. % Degradation means the percentage of the tube wall thickness affected or removed by degradation.

TABLE 3.6-1 (Continued)

CONTAINMENT ISOLATION VALVES

PENETRATION NO.	ISOLATION CHANNEL	ISOLATION VALVE IDENTIFICATION NO.	FUNCTION	ISOLATION TIME (SECONDS)
61	NA	SFP-176	Refueling Pool Outlet	NA
	NA	SFP-174		NA
	NA	SFP-172		NA
	NA	SFP-189		NA
62	SIAS A	PH-6579-MOV	Containment Heating Outlet	≤ 13
64	NA	PH-376	Containment Heating Outlet	NA

(1) Manual or remote manual valve which is closed during plant operation.

(2) May be opened below 300°F to establish shutdown cooling flow.

(3) Containment purge and containment vent isolation valves will be shut in MODES 1, 2, 3 and 4 per TS 3/4 6.1.7 and TS 3/4 6.1.8, respectively.

* May be open on an intermittent basis under administrative control.

** Containment purge isolation valves isolation times will only apply in MODE 6 when the valves are required to be OPERABLE and they are open. Isolation time for containment purge and containment vent isolation valves is NA for MODES 1, 2, 3 and 4 per TS 3/4 6.1.7 and TS 3/4 6.1.8, respectively, during which time these valves must remain closed.

3/4.6.5 COMBUSTIBLE GAS CONTROL

HYDROGEN ANALYZERS

LIMITING CONDITION FOR OPERATION

3.6.5.1 Two independent containment hydrogen analyzers shall be OPERABLE.

APPLICABILITY: MODES 1 and 2.

ACTION:

a. With one hydrogen analyzer inoperable, restore the inoperable analyzer to OPERABLE status within 30 days or be in at least HOT STANDBY within the next 6 hours.

b. With both hydrogen analyzers inoperable, restore at least one inoperable analyzer to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours.

SURVEILLANCE REQUIREMENTS

4.6.5.1 Each hydrogen analyzer shall be demonstrated OPERABLE at least bi-weekly on a STAGGERED TEST BASIS by drawing a sample from the waste gas system through the hydrogen analyzer.

4.6.5.2 Each hydrogen analyzer shall be demonstrated OPERABLE at least once per 92 days on a STAGGERED TEST BASIS by performing a CHANNEL CALIBRATION using sample gases in accordance with manufacturers' recommendations.

TABLE 3.7-5

FIRE PROTECTION SPRINKLERS

UNIT 1

SPRINKLER LOCATION	CONTROL VALVE ELEVATION
11 Diesel Generator	45'-0"
12 Diesel Generator	45'-0"
Unit 1 East Pipe Pen Room 227/316*	5'-0"
Unit 1 Aux Feed Pump Room 603*	12'-0"
Unit 1 East Piping Area Room 428*	45'-0"
Unit 1 East Electrical Penetration Room 429*	45'-0"
Unit 1 West Electrical Penetration Room 423*	45'-0"
Unit 1 Main Steam Piping Room 315*	45'-0"
Unit 1 Component Cooling Pump Room 228*	5'-0"
Unit 1 East Piping Area 224*	5'-0"
Unit 1 Radiation Exhaust Vent Equipment Room 225*	5'-0"
Unit 1 Service Water Pump Room 226*	5'-0"
Unit 1 Boric Acid Tank and Pump Room 217*	5'-0"
Unit 1 Reactor Coolant Makeup Pump Room 216*	5'-0"
Unit 1 Charging Pump Room 115*	(-)10'-0"
Unit 1 Misc Waste Mon Room 113*	(-)10'-0"
Cask and Eqpt Loading Area Rooms 419, 420, 425 & 426*	45'-0"
Solid Waste Processing*	45'-0"
Corridors 200, 202, 212 and 219*	5'-0"
Corridors 100, 103 and 116*	(-)10'-0"
Cable Chase 1A*	45'-0"
Cable Chase 1B*	45'-0"
Unit 1 ECCS Pump Room 119*	(-)15'-0"
Hot Instrument Shop Room 222*	5'-0"
Hot Machine Shop Room 223*	5'-0"

* Sprinklers required to ensure the OPERABILITY of redundant safe shutdown equipment.

HALON SYSTEMS

LIMITING CONDITION FOR OPERATION

3.7.11.3. The following Halon systems shall be OPERABLE with the storage tanks having at least 95% of full charge weight (or level) and 90% of full charge pressure.

a. Cable spreading rooms total flood system, and associated vertical cable chase 1C, Unit 1.

b. 4160 volt switchgear rooms 27 & 45' elevation Unit 1.

APPLICABILITY: Whenever equipment protected by the Halon system is required to be OPERABLE.

ACTION:

a. With both the primary and backup Halon systems protecting the areas inoperable, within one hour establish an hourly fire watch with backup fire suppression equipment for those areas protected by the inoperable Halon system. Restore the system to OPERABLE status within 14 days or prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within the next 30 days outlining the action taken, the cause of the inoperability and the plans and schedule for restoring the system to OPERABLE status.

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

SURVEILLANCE REQUIREMENTS

4.7.11.3 Each of the above required Halon systems shall be demonstrated OPERABLE:

a. At least once per 31 days by verifying that each valve (manual, power operated or automatic) in the flow path is in its correct position.

b. At least once per 6 months by verifying Halon storage tank weight (level) and pressure.

c. At least once per 12 months by performing a visual inspection of the nozzle(s) and visible flow paths for obstructions.

d. At least once per 18 months by verifying the system, including associated ventilation dampers and fire door release mechanisms, actuates manually and automatically, upon receipt of a simulated actuation signal, and

e. Following completion of major maintenance or modifications on the system(s), within 72 hours by performance of a flow test through headers and nozzles to assure no blockage.

DECAY TIME

LIMITING CONDITION FOR OPERATION

3.9.3 The reactor shall be subcritical for at least 72 hours.

APPLICABILITY: During movement of irradiated fuel in the reactor pressure vessel.

ACTION:

With the reactor subcritical for less than 72 hours, suspend all operations involving movement of irradiated fuel in the reactor pressure vessel. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.3 The reactor shall be determined to have been subcritical for at least 72 hours by verification of the date and time of subcriticality prior to movement of irradiated fuel in the reactor pressure vessel.

CONTAINMENT PENETRATIONS

LIMITING CONDITION FOR OPERATION

3.9.4 The containment penetrations shall be in the following status:

a. The equipment door closed and held in place by a minimum of four bolts,

b. A minimum of one door in each airlock is closed,* and

c. Each penetration providing direct access from the containment atmosphere to the outside atmosphere shall be either:

1. Closed by an isolation valve, blind flange, or manual valve, or

2. Be capable of being closed by an OPERABLE automatic containment purge valve.

APPLICABILITY: During CORE ALTERATIONS or movement of irradiated fuel within the containment.

ACTION:

With the requirements of the above specification not satisfied, immediately suspend all operations involving CORE ALTERATIONS or movement of irradiated fuel in the containment. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.4 Each of the above required containment penetrations shall be determined to be either in its closed/isolated condition or capable of being closed by an OPERABLE automatic containment purge valve within 72 hours prior to the start of and at least once per 7 days during CORE ALTERATIONS or movement of irradiated fuel in the containment by:

a. Verifying the penetrations are in their closed/isolated condition, or

b. Testing the containment purge valves per the applicable portions of Specification 4.6.4.1.2.

---

* The emergency escape hatch temporary closure device is an acceptable replacement for that airlock door.

SPENT FUEL POOL WATER LEVEL

LIMITING CONDITION FOR OPERATION

3.9.11 As a minimum, 21½ feet of water shall be maintained over the top of irradiated fuel assemblies seated in the storage racks.

APPLICABILITY: Whenever irradiated fuel assemblies are in the spent fuel pool.

ACTION:

With the requirement of the specification not satisfied, suspend all movement of fuel assemblies and crane operations with loads in the fuel storage areas and restore the water level to within its limit within 4 hours. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.11 The water level in the spent fuel pool shall be determined to be at least its minimum required depth at least once per 7 days when irradiated fuel assemblies are in the spent fuel pool.

SPENT FUEL POOL VENTILATION SYSTEM

LIMITING CONDITION FOR OPERATION

3.9.12 The spent fuel pool ventilation system shall be OPERABLE with:

a. One HEPA filter bank,

b. Two charcoal adsorber banks, and

c. Two exhaust fans.

APPLICABILITY: Whenever irradiated fuel is in the storage pool.

ACTION:

a. With one charcoal adsorber bank and/or one exhaust fan inoperable, fuel movement within the storage pool or crane operation with loads over the storage pool may proceed provided an OPERABLE exhaust fan is in operation and discharging through an OPERABLE train of HEPA filters and charcoal adsorbers.

b. With the HEPA filter bank inoperable, or with two charcoal adsorber banks inoperable, or with two exhaust fans inoperable, suspend all operations involving movement of fuel within the storage pool or crane operation with loads over the storage pool until at least one charcoal adsorber bank, at least one exhaust fan, and the HEPA filter bank are restored to OPERABLE status.

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

SURVEILLANCE REQUIREMENTS

4.9.12 The above required spent fuel pool ventilation system shall be demonstrated OPERABLE:

a. At least once per 31 days by initiating flow through the HEPA filter bank and both charcoal adsorber banks and verifying that each charcoal adsorber bank and each exhaust fan operates for at least 15 minutes.

SPECIAL TEST EXCEPTIONS

COOLANT CIRCULATION

LIMITING CONDITION FOR OPERATION

3.10.5 The reactor coolant circulation requirements of Specification 3.4.1 may be suspended and all reactor coolant pumps and shutdown cooling pumps may be de-energized during the time intervals required 1) for local leak rate testing of containment penetration number 41 pursuant to the requirements of Specification 4.6.1.2.d and 2) to permit maintenance on valves located in the common shutdown cooling suction line or on the shutdown cooling flow control valve (CV-306) provided:

a. No operations are permitted which could cause dilution of the reactor coolant system boron concentration, and specifically, the charging pumps shall be de-energized and the charging flow paths shall be closed,

b. The xenon reactivity is $\leq$ 0.1% $\Delta k/k$ and is approaching stability, and

c. The SHUTDOWN MARGIN requirement of Specification 3.1.1.2 is verified at least once per 8 hours when no shutdown cooling or reactor coolant pumps are in operation.

APPLICABILITY: MODES 4 and 5.

ACTION:

With the requirements of the above specification not satisfied, suspend all operations involving local leak rate testing of containment penetration number 41, maintenance on valves located in the common shutdown cooling suction line, and maintenance on valve CV-306.

SURVEILLANCE REQUIREMENTS

4.10.5.1 The charging pumps shall be verified de-energized and the charging flow paths shall be verified closed at least once per hour.

4.10.5.2 The xenon reactivity shall be determined to be $\leq$ 0.1% $\Delta k/k$ and approaching stability within 1 hour prior to suspending reactor coolant circulation.

The boron capability required below 200°F is based upon providing a 3% Δk/k SHUTDOWN MARGIN after xenon decay and cooldown from 200°F to 140°F. This condition requires either 737 gallons of 7.25% boric acid solution from the boric acid tanks or 9,844 gallons of 2300 ppm borated water from the refueling water tank.

The OPERABILITY of one boron injection system during REFUELING ensures that this system is available for reactivity control while in MODE 6.

3/4.1.3 MOVABLE CONTROL ASSEMBLIES

The specifications of this section ensure that (1) acceptable power distribution limits are maintained, (2) the minimum SHUTDOWN MARGIN is maintained, and (3) the potential effects of a CEA ejection accident are limited to acceptable levels.

The ACTION statements which permit limited variations from the basic requirements are accompanied by additional restrictions which ensure that the original criteria are met.

The ACTION statements applicable to a stuck or untrippable CEA and to a large misalignment (≥ 15 inches) of two or more CEAs, require a prompt shutdown of the reactor since either of these conditions may be indicative of a possible loss of mechanical functional capability of the CEAs and in the event of a stuck or untrippable CEA, the loss of SHUTDOWN MARGIN.

For small misalignments (≤ 15 inches) of the CEAs, there is 1) a small degradation in the peaking factors relative to those assumed in generating LCOs and LSSS setpoints for DNBR and linear heat rate, 2) a small effect on the time dependent long term power distributions relative to those used in generating LCOs and LSSS setpoints for DNBR and linear heat rate, 3) a small effect on the available SHUTDOWN MARGIN, and 4) a small effect on the ejected CEA worth used in the safety analysis. Therefore, the ACTION statement associated with the small misalignment of a CEA permits a one hour time interval during which attempts may be made to restore the CEA to within its alignment requirements prior to initiating a reduction in THERMAL POWER. The one hour time limit is sufficient to (1) identify causes of a misaligned CEA, (2) take appropriate corrective action to realign the CEAs and (3) minimize the effects of xenon redistribution.

Overpower margin is provided to protect the core in the event of a large misalignment (≥ 15 inches) of a CEA. However, this misalignment would cause distortion of the core power distribution. The reactor protective system would not detect the degradation in radial peaking factors and since variations in other system parameters (e.g., pressure and coolant temperature) may not be sufficient to cause trips, it is probable that the reactor could be operating with process variables less conservative than those assumed in generating LCO and LSSS setpoints. Therefore, the ACTION statement associated with the large misalignment of a CEA requires a prompt and significant reduction in THERMAL POWER prior to attempting realignment of the misaligned CEA.

The ACTION statements applicable to misaligned or inoperable CEAs include requirements to align the OPERABLE CEAs in a given group with the inoperable CEA. Conformance with these alignment requirements bring the core, within a short period of time, to a configuration consistent with that assumed in generating LCO and LSSS setpoints. However, extended operation with CEAs significantly inserted in the core may lead to perturbations in 1) local burnup, 2) peaking factors, and 3) available shutdown margin which are more adverse than the conditions assumed to exist in the safety analyses and LCO and LSSS setpoints determination. Therefore, time limits have been imposed on operation with inoperable CEAs to preclude such adverse conditions from developing.

Operability of the CEA position indicators is required to determine CEA positions and thereby ensure compliance with the CEA alignment and insertion limits and ensures proper operation of the rod block circuit. The CEA "Full In" and "Full Out" limits provide an additional independent means for determining the CEA positions when the CEAs are at either their fully inserted or fully withdrawn positions. Therefore, the OPERABILITY and the ACTION statements applicable to inoperable CEA position indicators permit continued operations when positions of CEAs with inoperable position indicators can be verified by the "Full In" or "Full Out" limits.

CEA positions and OPERABILITY of the CEA position indicators are required to be verified on a nominal basis of once per 12 hours with more frequent verifications required if an automatic monitoring channel is inoperable. These verification frequencies are adequate for assuring that the applicable LCOs are satisfied.

The surveillance requirements affecting CEAs with inoperable position indication channels allow 10 minutes for testing each affected CEA. This time limit was selected so that 1) the time would be long enough for the required testing, and 2) if all position indication were lost during testing, the time would be short enough to allow a power reduction to 70% of maximum allowable thermal power within one hour from when the testing was initiated. The time limit ensures CEA misalignments occurring during CEA testing are corrected within the time requirements required by existing specifications.

3/4.9.10 and 3/4.9.11 WATER LEVEL-REACTOR VESSEL AND SPENT FUEL POOL WATER LEVEL

The restrictions on minimum water level ensure that sufficient water depth is available to remove 99% of the assumed 10% iodine gap activity released from the rupture of an irradiated fuel assembly. The minimum water depth is consistent with the assumptions of the accident analysis.

3/4.9.12 SPENT FUEL POOL VENTILATION SYSTEM

The limitations on the spent fuel pool ventilation system ensure that all radioactive material released from an irradiated fuel assembly will be filtered through the HEPA filters and charcoal adsorber prior to discharge to the atmosphere. The OPERABILITY of this system and the resulting iodine removal capacity are consistent with the assumptions of the accident analyses.

3/4.9.13 SPENT FUEL CASK HANDLING CRANE

The restriction on movement of the spent fuel shipping cask within one cask length of any fuel assembly ensures that in the event this load is dropped (1) the stored spent fuel assemblies will not be damaged, and (2) any possible distortion of fuel in the storage racks will not result in a critical array.

3/4.9.14 CONTAINMENT VENT ISOLATION VALVES

The OPERABILITY and closure restrictions on the containment vent isolation valves 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.

6.7 SAFETY LIMIT VIOLATION

6.7.1 The following actions shall be taken in the event a Safety Limit is violated:

a. The facility shall be placed in at least HOT STANDBY within one hour.

b. The NRC Operations Center shall be notified by telephone as soon as possible and in all cases within one hour. The Manager - Nuclear Power Department and the OSSRC shall be notified within 24 hours.

c. A Safety Limit Violation Report shall be prepared. The report shall be reviewed by the POSRC. This report shall describe (1) applicable circumstances preceding the violation, (2) effects of the violation upon facility components, systems or structures, and (3) corrective action taken to prevent recurrence.

d. The Safety Limit Violation Report shall be submitted to the Commission, the OSSRC and the Manager - Nuclear Power Department within 14 days of the violation.

6.8 PROCEDURES

6.8.1 Written procedures shall be established, implemented and maintained covering the activities referenced below:

a. The applicable procedures recommended in Appendix "A" of Regulatory Guide 1.33, Revision 2, February 1978.

b. Refueling operations.

c. Surveillance and test activities of safety related equipment.

d. Security Plan implementation.

e. Emergency Plan implementation.

f. Fire Protection Program implementation.

g. The amount of overtime worked by plant staff members performing safety related functions must be limited in accordance with the NRC Policy Statement on working Hours (Generic Letter No. 82-12).

h. PROCESS CONTROL PROGRAM implementation.

i. OFFSITE DOSE CALCULATION MANUAL implementation.

6.8.2 Each procedure and administrative policy of 6.8.1 above and changes thereto shall be reviewed by the POSRC and approved by the Plant Superintendent prior to implementation and reviewed periodically as set forth in administrative procedures.

6.8.3 Temporary changes to procedures of 6.8.1 above may be made provided:

a. The intent of the original procedure is not altered.

b. The change is approved by two members of the plant management staff, at least one of whom holds a Senior Reactor Operator's License on the unit affected.

c. The change is documented, reviewed by the POSRC and approved by the Plant Superintendent within 14 days of implementation.

6.9 REPORTING REQUIREMENTS

ROUTINE REPORTS

6.9.1 In addition to the applicable reporting requirements of Title 10, Code of Federal Regulations, the following reports shall be submitted to the Director of the Regional Office of Inspection and Enforcement unless otherwise noted.

STARTUP REPORT

6.9.1.1 A summary report of plant startup and power escalation testing shall be submitted following (1) receipt of an operating license, (2) amendment to the license involving a planned increase in power level, (3) installation of fuel that has a different design or has been manufactured by a different fuel supplier, and (4) modifications that may have significantly altered the nuclear, thermal, or hydraulic performance of the plant.

6.9.1.2 The startup report shall address each of the tests identified in the FSAR and shall include a description of the measured values of the operating conditions or characteristics obtained during the test program and a comparison of these values with design predictions and specifications. Any corrective actions that were required to obtain satisfactory operation shall also be described. Any additional specific details required in license conditions based on other commitments shall be included in this report.

b. A high radiation area in which the intensity of radiation is greater than 1000 mrem/hr shall be subject to the provisions of 6.12.1.a, above, and in addition locked barricades shall be provided to prevent unauthorized entry into such areas and the keys shall be maintained by the Supervisor-Radiation Control and the Operations Shift Supervisor on duty under their separate administrative control.

6.13 SYSTEM INTEGRITY

The licensee shall implement a program to reduce leakage from systems outside containment that would or could contain highly radioactive fluids during a serious transient or accident to as low as practical levels. This program shall include the following:

1. Provisions establishing preventive maintenance and periodic visual inspection requirements, and

2. Leak test requirements for each system at a frequency not to exceed refueling cycle intervals.

6.14 IODINE MONITORING

The licensee shall implement a program which will ensure the capability to accurately determine the airborne iodine concentration in vital areas under accident conditions. This program shall include the following:

1. Training of personnel,

2. Procedures for monitoring, and

3. Provisions for maintenance of sampling and analysis equipment.

6.15 PROCESS CONTROL PROGRAM (PCP)

6.15.1 The PCP shall be approved by the Commission prior to implementation.

6.15.2 Licensee initiated changes to the PCP:

1. Shall be submitted to the Commission in the Semiannual Radioactive Effluent Release Report for the period in which the change(s) was made. This submittal shall contain:

   a. An evaluation supporting the premise that the change did not reduce the overall conformance of the solidified waste product to existing criteria for solid wastes; and

b. A reference to the date and the POSRC meeting number in which the change(s) were reviewed and found acceptable to the POSRC.

2. Shall become effective upon review and approval by the responsible Nuclear Power Department unit and approval of Plant Superintendent.

6.16 OFFSITE DOSE CALCULATION MANUAL (ODCM)

6.16.1 The ODCM shall be approved by the Commission prior to implementation.

6.16.2 Licensee initiated changes to the ODCM:

1. Shall be submitted to the Commission in the Semiannual Radioactive Effluent Release Report for the period in which the change(s) was made effective. This submittal shall contain:

   a. Sufficient information to support the rationale for the change. Information submitted should consist of a package of those pages of the ODCM to be changed with each page numbered and provided with a change number and/or change date together with appropriate analyses or evaluations justifying the change(s);

   b. A determination that the change will not reduce the accuracy or reliability of dose calculations or setpoint determinations; and

   c. Documentation of the fact that the change has been reviewed and found acceptable by the POSRC.

2. Shall become effective upon review by the POSRC and approval of the Plant Superintendent.

6.17 MAJOR CHANGES TO RADIOACTIVE LIQUID, GASEOUS AND SOLID WASTE TREATMENT SYSTEMS

6.17.1 Licensee initiated major changes to the radioactive waste systems (liquid, gaseous and solid) shall be reported to the Commission in the Semi-annual Radioactive Effluent Release Report for the period in which the modification to the waste system is completed. The discussion of each change shall contain:

a. A description of the equipment, components and processes involved.

b. Documentation of the fact that the change including the safety analysis was reviewed and found acceptable by the POSRC.

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LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

ATTACHMENT TO LICENSE AMENDMENT NO. 91

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
VI	VI
XVI	XVI
3/4 1-21	3/4 1-21
3/4 1-25	3/4 1-25
3/4 3-13	3/4 3-13
3/4 3-16	3/4 3-16
3/4 3-41	3/4 3-41
3/4 4-8	3/4 4-8
3/4 6-25	3/4 6-25
3/4 7-64	3/4 7-64
3/4 9-4	3/4 9-4
3/4 9-11	3/4 9-11
3/4 10-5	3/4 10-5
B 3/4 9-3	B 3/4 9-3
6-14	6-14
6-21	6-21
6-22	6-22

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. 91
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 June 28, 1985 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. 91 , 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 4, 1985

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ADMINISTRATIVE CONTROLS

INDEX

ADMINISTRATIVE CONTROLS

POSITION INDICATOR CHANNELS

LIMITING CONDITION FOR OPERATION

3.1.3.3 At least two of the following three CEA position indicator channels shall be OPERABLE for each shutdown and regulating CEA:

a. CEA voltage divider reed switch position indicator channel, capable of determining the absolute CEA position within ± 1.75 inches;

b. CEA "Full Out" or "Full In" reed switch position indicator channel, only if the CEA is fully withdrawn or fully inserted, as verified by actuation of the applicable position indicator; and

c. CEA pulse counting position indicator channel.

APPLICABILITY: MODES 1 and 2.

ACTION:

a. With a maximum of one CEA per group having its voltage divider reed switch position indicator channel or its pulse counting position indicator channel inoperable and the CEA(s) with the inoperable position indicator channel partially inserted, either:

1. Within 6 hours

a) Restore the inoperable position indicator channel to OPERABLE status, or

b) Be in at least HOT STANDBY, or

c) Reduce THERMAL POWER to < 70% of the maximum allowable THERMAL POWER level for the existing Reactor Coolant Pump combination; if negative reactivity insertion is required to reduce THERMAL POWER, boration shall be used. Operation at or below this reduced THERMAL POWER level may continue provided that within the next 4 hours either:

1) The CEA group(s) with the inoperable position indicator is fully withdrawn while maintaining the withdrawal sequence required by Specification 3.1.3.6 and when this CEA group reaches its fully withdrawn position, the "Full Out" limit of the CEA with the inoperable position indicator is actuated and verifies this CEA to be fully withdrawn. Subsequent to fully withdrawing this CEA group(s), the THERMAL POWER level may be returned to a level consistent with all other applicable specifications and operation may continue per Specification 3.1.3.3 above; or

2) The CEA group(s) with the inoperable position indicator is fully inserted, and subsequently maintained fully inserted, while maintaining the withdrawal sequence and THERMAL POWER level required by Specification 3.1.3.6 and when this CEA group reaches its fully

LIMITING CONDITION FOR OPERATION

inserted position, the "Full In" limit of the CEA with the inoperable indicator is actuated and verifies this CEA to be fully inserted. Subsequent operation shall be within the limits of Specification 3.1.3.6, and may continue per Specification 3.1.3.3 above.

2. or, if the failure existed before entry into MODE 2 or occurs prior to an "all CEAs out" configuration, the CEA group(s) with inoperable position indicator channel must be moved to the "Full Out" position and verified to be fully withdrawn via a "Full Out" indicator. These actions must be completed within 10 hours of entry into MODE 2 and prior to exceeding 70% of the maximum allowable THERMAL POWER level for the existing Reactor Coolant Pump combination. The provisions of Specification 3.0.4 are not applicable. Once these actions are completed, operation may continue per Specification 3.1.3.3 above.

b. With more than one CEA per group having its CEA pulse counting position indicator channel and either (1) the "Full Out" or "Full In" position indicator, or (2) the voltage divider position indicator channel inoperable, operation in MODES 1 and 2 may continue for up to 24 hours provided that for the affected CEAs, either:

1. The CEA voltage divider reed switch position indicator channels are OPERABLE, or

2. The CEA "Full Out" or "Full In" reed switch position indicator channels are OPERABLE, with the CEA fully withdrawn or fully inserted as verified by actuation of the applicable position indicator.

SURVEILLANCE REQUIREMENTS

4.1.3.3.1 Each required CEA position indication channel shall be determined to be OPERABLE by determining CEA positions as follows at least once per 12 hours, by:

a. Verifying the CEA pulse counting position indicator channels and the CEA voltage divider reed switch position indicator channels agree within 4.5 inches, or

b. Verifying the CEA pulse counting position indicator channels and the CEA "Full Out" or Full In" reed switch position indicator channels agree within 4.5 inches, or

c. Verifying the CEA voltage divider reed switch position indicator channels and the CEA "Full Out" or "Full In" reed switch position indicator channels agree within 4.5 inches.

4.1.3.3.2 During time intervals when the deviation circuit is inoperable, the above verification of required CEA position indicator channels shall be made at least once per 4 hours.

REGULATING CEA INSERTION LIMITS

LIMITING CONDITION FOR OPERATION

3.1.3.6 The regulating CEA groups shall be limited to the withdrawal sequence and to the insertion limits shown on Figure 3.1-2 (regulating CEAs are considered to be fully withdrawn in accordance with Figure 3.1-2 when withdrawn to at least 129.0 inches) with CEA insertion between the Long Term Steady State Insertion Limits and the Transient Insertion Limits restricted to:

a. ≤ 4 hours per 24 hour interval,

b. ≤ 5 Effective Full Power Days per 30 Effective Full Power Day interval, and

c. ≤ 14 Effective Full Power Days per calendar year.

APPLICABILITY: MODES 1* and 2*#.

ACTION:

a. With the regulating CEA groups inserted beyond the Transient Insertion Limits, except for surveillance testing pursuant to Specification 4.1.3.1.2, within two hours either:

   1. Restore the regulating CEA groups to within the limits, or

   2. Reduce THERMAL POWER to less than or equal to that fraction of RATED THERMAL POWER which is allowed by the CEA group position using Figure 3.1-2.

b. With the regulating CEA groups inserted between the Long Term Steady State Insertion Limits and the Transient Insertion Limits for intervals > 4 hours per 24 hour interval, except during operations pursuant to the provisions of ACTION items c. and e. of Specification 3.1.3.1, operation may proceed provided either:

   1. The Short Term Steady State Insertion Limits of Figure 3.1-2 are not exceeded, or

   2. Any subsequent increase in THERMAL POWER is restricted to ≤ 5% of RATED THERMAL POWER per hour.

*See Special Test Exceptions 3.10.2 and 3.10.4.

#With $K_{eff} \geq 1.0$.

LIMITING CONDITION FOR OPERATION (Continued)

c. With the regulating CEA groups inserted between the Long Term Steady State Insertion Limits and the Transient Insertion Limits for intervals > 5 EFPD per 30 EFPD interval or > 14 EFPD per calendar year, except during operations pursuant to the provisions of ACTION items c. and e. of Specification 3.1.3.1, either:

1. Restore the regulating groups to within the Long Term Steady State Insertion Limits within two hours, or

2. Be in at least HOT STANDBY within 6 hours.

SURVEILLANCE REQUIREMENTS

4.1.3.6 The position of each regulating CEA group shall be determined to be within the Transient Insertion Limits at least once per 12 hours except during time intervals when the PDIL Alarm Circuit is inoperable, then verify the individual CEA positions at least once per 4 hours. The accumulated times during which the regulating CEA groups are inserted beyond the Steady State Insertion Limits but within the Transient Insertion Limits shall be determined at least once per 24 hours.

TABLE 3.3-3 (Continued)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
6. CONTAINMENT PURGE VALVES ISOLATION ##					
a. Manual (Purge Valve Control Switches)	2/Penetration	1/Penetration	2/Penetration	6*	8
b. Containment Radiation - High Area Monitor	4	2	3	6*	8
7. LOSS OF POWER					
a. 4.16 kv Emergency Bus Undervoltage (Loss of Voltage	4/Bus	2/Bus	3/Bus	1, 2, 3	7*
b. 4.16 kv Emergency Bus Undervoltage (Degraded Voltage)	4/Bus	2/Bus	3/Bus	1, 2, 3	7*

Containment purge valve isolation is also initiated by SIAS (functional units 1.a, 1.b, and 1.c).

* Must be OPERABLE only in MODE 6 when the valves are required OPERABLE and they are open.

TABLE 3.3-3 (Continued)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
8. CVCS ISOLATION					
a. Manual (CVCS Isolation Valve Control Switches)	1/Valve	1/Valve	1/Valve	1, 2, 3, 4	6
b. West Penetration Room/Letdown Heat Exchanger Room Pressure - High	4	2	3	1, 2, 3, 4	7*
9. AUXILIARY FEEDWATER					
a. Manual	2 sets of 2 per S/G	1 set of 2 per S/G	2 sets of 2 per S/G	1, 2, 3	6
b. Steam Generator Level - Low	4/SG	2/SG	3/SG	1, 2, 3	7
c. Steam Generator ΔP High	4/SG	2/SG	3/SG	1, 2, 3	7

TABLE 3.3-3 (Continued)

TABLE NOTATION

(a) Trip function may be bypassed in this MODE when pressurizer pressure is < 1800 psia; bypass shall be automatically removed when pressurizer pressure is ≥ 1800 psia.

(c) Trip function may be bypassed in this MODE below 785 psia; bypass shall be automatically removed at or above 785 psia.

* The provisions of Specification 3.0.4 are not applicable.

ACTION STATEMENTS

ACTION 6 - With the number of OPERABLE channels one less than the Total Number of Channels, restore the inoperable channel to OPERABLE status within 48 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

ACTION 7 - With the number of OPERABLE channels one less than the Total Number of Channels, operation may proceed provided the following conditions are satisfied:

a. The inoperable channel is placed in either the bypassed or tripped condition within 1 hour. For the purposes of testing and maintenance, the inoperable channel may be bypassed for up to 48 hours from time of initial loss of OPERABILITY; however, the inoperable channel shall then be either restored to OPERABLE status or placed in the tripped condition.

b. Within one hour, all functional units receiving an input from the inoperable channel are also placed in the same condition (either bypassed or tripped, as applicable) as that required by a. above for the inoperable channel.

c. The Minimum Channels OPERABLE requirement is met; however, one additional channel may be bypassed for up to 48 hours while performing tests and maintenance on that channel provided the other inoperable channel is placed in the tripped condition.

TABLE 3.3-3 (Continued)

ACTION 8 - With less than the Minimum Channels OPERABLE, operation may continue provided the containment purge valves are maintained closed.

ACTION 11 - With the number of OPERABLE Channels one less than the Total Number of Channels, operation may proceed provided the inoperable channel is placed in the bypassed condition and the Minimum Channels OPERABLE requirement is demonstrated within 1 hour; one additional channel may be bypassed for up to 2 hours for surveillance testing per Specification 4.3.2.1.

TABLE 3.3-10

POST-ACCIDENT MONITORING INSTRUMENTATION

INSTRUMENT	MINIMUM CHANNELS OPERABLE
1. Containment Pressure	2
2. Wide Range Logarithmic Neutron Flux Monitor	2
3. Reactor Coolant Outlet Temperature	2
4. Pressurizer Pressure	2
5. Pressurizer Level	2
6. Steam Generator Pressure	2/steam generator
7. Steam Generator Level (Wide Range)	2/steam generator
8. Auxiliary Feedwater Flow Rate	2/steam generator
9. RCS Subcooled Margin Monitor	1
10. PORV/Safety Valve Acoustic Flow Monitoring	1/valve
11. PORV Solenoid Power Indication	1/valve
12. Feedwater Flow	2
13. Containment Water Level (Wide Range)	1

TABLE 4.3-10

POST-ACCIDENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	CHANNEL CALIBRATION
1. Containment Pressure	M	R
2. Wide Range Logarithmic Neutron Flux Monitor	M	N.A.
3. Reactor Coolant Outlet Temperature	M	R
4. Pressurizer Pressure	M	R
5. Pressurizer Level	M	R
6. Steam Generator Pressure	M	R
7. Steam Generator Level (Wide Range)	M	R
8. Auxiliary Feedwater Flow Rate	M	R
9. RCS Subcooled Margin Monitor	M	R
10. PORV/Safety Valve Acoustic Monitor	N.A.	R
11. PORV Solenoid Power Indication	N.A.	N.A.
12. Feedwater Flow	M	R
13. Containment Water Level (Wide Range)	M	R

2. Tubes in those areas where experience has indicated potential problems.

c. The second and third inservice inspections may be less than a full tube inspection by concentrating (selecting at least 50% of the tubes to be inspected) the inspection on those areas of the tube sheet array and on those portions of the tubes where tubes with imperfections were previously found.

The results of each sample inspection shall be classified into one of the following three categories:

Category	Inspection Results
C-1	Less than 5% of the total tubes inspected are degraded tubes and none of the inspected tubes are defective.
C-2	One or more tubes, but not more than 1% of the total tubes inspected are defective, or between 5% and 10% of the total tubes inspected are degraded tubes.
C-3	More than 10% of the total tubes inspected are degraded tubes or more than 1% of the inspected tubes are defective.

Note: In all inspections, previously degraded tubes must exhibit significant (>10%) further wall penetrations to be included in the above percentage calculations.

4.4.5.3 Inspection Frequencies - The above required inservice inspections of steam generator tubes shall be performed at the following frequencies:

a. The first inservice inspection shall be performed after 6 Effective Full Power Months but within 24 calendar months of initial criticality. Subsequent inservice inspections shall be performed at intervals of not less than 12 nor more than 24 calendar months after the previous inspection. If two consecutive inspections following service under AVT conditions, not including the preservice inspection, result in all inspection results falling into the C-1 category or if two consecutive inspections demonstrate that previously observed degradation has not continued and no additional degradation has occurred, the inspection interval may be extended to a maximum of once per 40 months.

SURVEILLANCE REQUIREMENTS (Continued)

b. If the inservice inspection of a steam generator conducted in accordance with Table 4.4-2 requires a third sample inspection whose results fall in Category C-3, the inspection frequency shall be reduced to at least once per 20 months. The reduction in inspection frequency shall apply until a subsequent inspection demonstrates that a third sample inspection is not required.

c. Additional, unscheduled inservice inspections shall be performed on each steam generator in accordance with the first sample inspection specified in Table 4.4-2 during the shutdown subsequent to any of the following conditions:

1. Primary-to-secondary tube leaks (not including leaks originating from tube-to-tube sheet welds) in excess of the limits of Specification 3.4.6.2,

2. A seismic occurrence greater than the Operating Basis Earthquake,

3. A loss-of-coolant accident requiring actuation of the engineered safeguards, or

4. A main steam line or feedwater line break.

4.4.5.4 Acceptance Criteria

a. As used in this Specification:

1. Imperfection means an exception to the dimensions, finish or contour of a tube from that required by fabrication drawings or specifications. Eddy-current testing indications below 20% of the nominal tube wall thickness, if detectable, may be considered as imperfections.

2. Degradation means a service-induced cracking, wastage, wear or general corrosion occurring on either inside or outside of a tube.

3. Degraded Tube means a tube containing imperfections $\geq$20% of the nominal wall thickness caused by degradation.

4. % Degradation means the percentage of the tube wall thickness affected or removed by degradation.

TABLE 3.6-1 (Continued)

CONTAINMENT ISOLATION VALVES

PENETRATION NO.	ISOLATION CHANNEL	ISOLATION VALVE IDENTIFICATION NO.	FUNCTION	ISOLATION TIME (SECONDS)
61	NA	SFP-184	Refueling Pool Outlet	NA
	NA	SFP-182		NA
	NA	SFP-180		NA
	NA	SFP-186		NA
62	SIAS A	PH-6579-MOV	Containment Heating Outlet	≤13
64	NA	PH-387	Containment Heating Inlet	NA

(1) Manual or remote manual valve which is closed during plant operation.

(2) May be opened below 300°F to establish shutdown cooling flow.

(3) Containment purge and containment vent isolation valves will be shut in MODES 1, 2, 3 and 4 per TS 3/4 6.1.7 and TS 3/4 6.1.8, respectively.

* May be open on an intermittent basis under administrative control.

** Containment purge isolation valves isolation times will only apply in MODE 6 when the valves are required to be OPERABLE and they are open. Isolation time for containment purge and containment vent isolation valves is NA for MODES 1, 2, 3 and 4 per TS 3/4 6.1.7 and TS 3/4 6.1.8, respectively, during which time these valves must remain closed.

3/4.6.5 COMBUSTIBLE GAS CONTROL

HYDROGEN ANALYZERS

LIMITING CONDITION FOR OPERATION

3.6.5.1 Two independent containment hydrogen analyzers shall be OPERABLE.

APPLICABILITY: MODES 1 and 2.

ACTION:

a. With one hydrogen analyzer inoperable, restore the inoperable analyzer to OPERABLE status within 30 days or be in at least HOT STANDBY within the next 6 hours.

b. With both hydrogen analyzers inoperable, restore at least one inoperable analyzer to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours.

SURVEILLANCE REQUIREMENTS

4.6.5.1 Each hydrogen analyzer shall be demonstrated OPERABLE at least biweekly on a STAGGERED TEST BASIS by drawing a sample from the Waste Gas System through the hydrogen analyzer indicator.

4.6.5.2 Each hydrogen analyzer shall be demonstrated OPERABLE at least once per 92 days on a STAGGERED TEST BASIS by performing a CHANNEL CALIBRATION using sample gases in accordance with manufacturers' recommendations.

SPENT FUEL POOL WATER LEVEL

LIMITING CONDITION FOR OPERATION

3.9.11 As a minimum, 21½ feet of water shall be maintained over the top of irradiated fuel assemblies seated in the storage racks.

APPLICABILITY: Whenever irradiated fuel assemblies are in the spent fuel pool.

ACTION:

With the requirement of the specification not satisfied, suspend all movement of fuel assemblies and crane operations with loads in the fuel storage areas and restore the water level to within its limit within 4 hours. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.11 The water level in the spent fuel pool shall be determined to be at least its minimum required depth at least once per 7 days when irradiated fuel assemblies are in the spent fuel pool.

SPENT FUEL POOL VENTILATION SYSTEM

LIMITING CONDITION FOR OPERATION

3.9.12 The spent fuel pool ventilation system shall be OPERABLE with:

a. One HEPA filter bank,

b. Two charcoal adsorber banks, and

c. Two exhaust fans.

APPLICABILITY: Whenever irradiated fuel is in the storage pool.

ACTION:

a. With one charcoal adsorber bank and/or one exhaust fan inoperable, fuel movement within the storage pool or crane operation with loads over the storage pool may proceed provided an OPERABLE exhaust fan is in operation and discharging through an OPERABLE train of HEPA filters and charcoal adsorbers.

b. With the HEPA filter bank inoperable, or with two charcoal adsorber banks inoperable, or with two exhaust fans inoperable, suspend all operations involving movement of fuel within the storage pool or crane operation with loads over the storage pool until at least one charcoal adsorber bank, at least one exhaust fan, and the HEPA filter bank are restored to OPERABLE status.

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

SURVEILLANCE REQUIREMENTS

4.9.12 The above required spent fuel pool ventilation system shall be demonstrated OPERABLE:

a. At least once per 31 days by initiating flow through the HEPA filter bank and both charcoal adsorber banks and verifying that each charcoal adsorber bank and each exhaust fan operates for at least 15 minutes.

TABLE 3.7-5

FIRE PROTECTION SPRINKLERS

UNIT 2

SPRINKLER LOCATION	CONTROL VALVE ELEVATION
Unit 2 Aux Feed Pump Room 605*	12'-0"
Unit 2 East Piping Area Room 408*	45'-0"
Unit 2 East Elec Pen Room 409*	45'-0"
Unit 2 West Elec Pen Room 414*	45'-0"
Cable Chase 2A*	45'-0"
Cable Chase 2B*	45'-0"
Unit 2 Main Steam Piping Room 309*	45'-0"
Unit 2 Component Cooling Pp Room 201	5'-0"
Unit 2 East Piping Area 203*	5'-0"
Unit 2 Rad Exh Vent Equip Room 204*	5'-0"
Unit 2 Service Water Pp Room 205*	5'-0"
Unit 2 Boric Acid TK and Pp Room 215*	5'-0"
Unit 2 Reactor Coolant Makeup Pump Room 216A*	5'-0"
Unit 2 Charging Pump Room 105*	(-)10'-0"
Unit 2 Misc Waste Monitor TK Room 106*	(-)10'-0"
Unit 2 ECCS Pump Room 101*	(-)15'-0"
21 Diesel Generator	45'-0"
Unit 2 East Pipe Pen Room 206/310*	5'-0"

NOTE: Sprinklers protecting all rooms listed under heading "Unit 2" will be made operational later in 1981 except for "21 Diesel Generator" which is now operational.

*Sprinklers required to ensure the OPERABILITY of redundant safe shutdown equipment.

HALON SYSTEMS

LIMITING CONDITION FOR OPERATION

3.7.11.3 The following Halon systems shall be OPERABLE with the storage tanks having at least 95% of full charge weight (or level) and 90% of full charge pressure.

a. Cable spreading rooms total flood system, and associated vertical cable chase 1C, Unit 2.

b. 4160 volt switchgear rooms 27 & 45" elevation Unit 2.

APPLICABILITY: Whenever equipment protected by the Halon system is required to be OPERABLE.

ACTION:

a. With both the primary and backup Halon systems protecting the areas inoperable, within one hour establish an hourly fire watch with backup fire suppression equipment for those areas protected by the inoperable Halon system. Restore the system to OPERABLE status within 14 days or prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within the next 30 days outlining the action taken, the cause of the inoperability and the plans and schedule for restoring the system to OPERABLE status.

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

SURVEILLANCE REQUIREMENTS

4.7.11.3 Each of the above required Halon systems shall be demonstrated OPERABLE:

a. At least once per 31 days by verifying that each valve (manual, power operated or automatic) in the flow path is in its correct position.

b. At least once per 6 months by verifying Halon storage tank weight (level) and pressure.

c. At least once per 12 months by performing a visual inspection of the nozzle(s) and visible flow paths for obstructions.

d. At least once per 18 months by verifying the system, including associated ventilation dampers and fire door release mechanisms, actuates manually and automatically, upon receipt of a simulated actuation signal, and

e. Following completion of major maintenance or modifications on the system(s), within 72 hours by performance of a flow test through headers and nozzles to assure no blockage.

DECAY TIME

LIMITING CONDITION FOR OPERATION

3.9.3 The reactor shall be subcritical for at least 72 hours.

APPLICABILITY: During movement of irradiated fuel in the reactor pressure vessel.

ACTION:

With the reactor subcritical for less than 72 hours, suspend all operations involving movement of irradiated fuel in the reactor pressure vessel. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.3 The reactor shall be determined to have been subcritical for at least 72 hours by verification of the date and time of subcriticality prior to movement of irradiated fuel in the reactor pressure vessel.

CONTAINMENT PENETRATIONS

LIMITING CONDITION FOR OPERATION

3.9.4 The containment penetrations shall be in the following status:

a. The equipment door closed and held in place by a minimum of four bolts,

b. A minimum of one door in each airlock is closed*, and

c. Each penetration providing direct access from the containment atmosphere to the outside atmosphere shall be either:

1. Closed by an isolation valve, blind flange, or manual valve, or

2. Be capable of being closed by an OPERABLE automatic containment purge valve.

APPLICABILITY: During CORE ALTERATIONS or movement of irradiated fuel within the containment.

ACTION:

With the requirements of the above specification not satisfied, immediately suspend all operations involving CORE ALTERATIONS or movement of irradiated fuel in the containment. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.4 Each of the above required containment penetrations shall be determined to be either in its closed/isolated condition or capable of being closed by an OPERABLE automatic containment purge valve within 72 hours prior to the start of and at least once per 7 days during CORE ALTERATIONS or movement of irradiated fuel in the containment by:

a. Verifying the penetrations are in their closed/isolated condition, or

b. Testing the containment purge valves per the applicable portions of Specification 4.6.4.1.2.

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* The emergency escape hatch temporary closure device is an acceptable replacement for that airlock door.

SPECIAL TEST EXCEPTIONS

COOLANT CIRCULATION

LIMITING CONDITION FOR OPERATION

3.10.5 The reactor coolant circulation requirements of Specification 3.4.1 may be suspended and all reactor coolant pumps and shutdown cooling pumps may be de-energized during the time intervals required 1) for local leak rate testing of containment penetration number 41 pursuant to the requirements of Specification 4.6.1.2.d and 2) to permit maintenance on valves located in the common shutdown cooling suction line or on the shutdown cooling flow control valve (CV-306) provided:

a. No operations are permitted which could cause dilution of the reactor coolant system boron concentration, and specifically, the charging pumps shall be de-energized and the charging flow paths shall be closed,

b. The xenon reactivity is $\leq$ 0.1% $\Delta k/k$ and is approaching stability, and

c. The SHUTDOWN MARGIN requirement of Specification 3.1.1.2 is verified at least once per 8 hours when no shutdown cooling or reactor coolant pumps are in operation.

APPLICABILITY: MODES 4 and 5.

ACTION:

With the requirements of the above specification not satisfied, suspend all operations involving local leak rate testing of containment penetration number 41, maintenance on valves located in the common shutdown cooling suction line, and maintenance on valve CV-306.

SURVEILLANCE REQUIREMENTS

4.10.5.1 The charging pumps shall be verified de-energized and the charging flow paths shall be verified closed at least once per hour.

4.10.5.2 The xenon reactivity shall be determined to be $\leq$ 0.1% $\Delta k/k$ and approaching stability within 1 hour prior to suspending reactor coolant circulation.

3/4.9.10 and 3/4.9.11 WATER LEVEL-REACTOR VESSEL AND SPENT FUEL POOL WATER LEVEL

The restrictions on minimum water level ensure that sufficient water depth is available to remove 99% of the assumed 10% iodine gap activity released from the rupture of an irradiated fuel assembly. The minimum water depth is consistent with the assumptions of the accident analysis.

3/4.9.12 SPENT FUEL POOL VENTILATION SYSTEM

The limitations on the spent fuel pool ventilation system ensure that all radioactive material released from an irradiated fuel assembly will be filtered through the HEPA filters and charcoal adsorber prior to discharge to the atmosphere. The OPERABILITY of this system and the resulting iodine removal capacity are consistent with the assumptions of the accident analyses.

3/4.9.13 SPENT FUEL CASK HANDLING CRANE

The restriction on movement of the spent fuel shipping cask within one cask length of any fuel assembly ensures that in the event this load is dropped (1) the stored spent fuel assemblies will not be damaged, and (2) any possible distortion of fuel in the storage racks will not result in a critical array.

3/4.9.14 CONTAINMENT VENT ISOLATION VALVES

The OPERABILITY and closure restrictions on the containment vent isolation valves 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.

6.8.3 Temporary changes to procedures of 6.8.1 above may be made provided:

a. The intent of the original procedure is not altered.

b. The change is approved by two members of the plant management staff, at least one of whom holds a Senior Reactor Operator's License on the unit affected.

c. The change is documented, reviewed by the POSRC and approved by the Plant Superintendent within 14 days of implementation.

6.9 REPORTING REQUIREMENTS

ROUTINE REPORTS

6.9.1 In addition to the applicable reporting requirements of Title 10, Code of Federal Regulations, the following reports shall be submitted to the Director of the Regional Office of Inspection and Enforcement unless otherwise noted.

STARTUP REPORT

6.9.1.1 A summary report of plant startup and power escalation testing shall be submitted following (1) receipt of an operating license, (2) amendment to the license involving a planned increase in power level, (3) installation of fuel that has a different design or has been manufactured by a different fuel supplier, and (4) modifications that may have significantly altered the nuclear, thermal, or hydraulic performance of the plant.

6.9.1.2 The startup report shall address each of the tests identified in the FSAR and shall include a description of the measured values of the operating conditions or characteristics obtained during the test program and a comparison of these values with design predictions and specifications. Any corrective actions that were required to obtain satisfactory operation shall also be described. Any additional specific details required in license conditions based on other commitments shall be included in this report.

b. A high radiation area in which the intensity of radiation is greater than 1000 mrem/hr shall be subject to the provisions of 6.12.1.a, above, and in addition locked barricades shall be provided to prevent unauthorized entry into such areas and the keys shall be maintained by the Supervisor-Radiation Control and the Operations Shift Supervisor on duty under their separate administrative control.

6.13 SYSTEM INTEGRITY

The licensee shall implement a program to reduce leakage from systems outside containment that would or could contain highly radioactive fluids during a serious transient or accident to as low as practical levels. This program shall include the following:

1. Provisions establishing preventive maintenance and periodic visual inspection requirements, and

2. Leak test requirements for each system at a frequency not to exceed refueling cycle intervals.

6.14 IODINE MONITORING

The licensee shall implement a program which will ensure the capability to accurately determine the airborne iodine concentration in vital areas under accident conditions. This program shall include the following:

1. Training of personnel,

2. Procedures for monitoring, and

3. Provisions for maintenance of sampling and analysis equipment.

6.15 PROCESS CONTROL PROGRAM (PCP)

6.15.1 The PCP shall be approved by the Commission prior to implementation.

6.15.2 Licensee initiated changes to the PCP:

1. Shall be submitted to the Commission in the Semiannaul Radioactive Effluent Release Report for the period in which the change(s) was made. This submittal shall contain:

a. An evaluation supporting the premise that the change did not reduce the overall conformance of the solidified waste product to existing criteria for solid wastes; and

b. A reference to the date and the POSRC meeting number in which the change(s) were reviewed and found acceptable to the POSRC.

2. Shall become effective upon review and approval by the responsible Nuclear Power Department unit and approval of Plant Superintendent.

6.16 OFFSITE DOSE CALCULATION MANUAL (ODCM)

6.16.1 The ODCM shall be approved by the Commission prior to implementation.

6.16.2 Licensee initiated changes to the ODCM:

1. Shall be submitted to the Commission in the Semiannual Radioactive Effluent Release Report for the period in which the change(s) was made effective. This submittal shall contain:

   a. Sufficient information to support the rationale for the change. Information submitted should consist of a package of those pages of the ODCM to be changed with each page numbered and provided with a change number and/or change date together with appropriate analyses or evaluations justifying the change(s);

   b. A determination that the change will not reduce the accuracy or reliability of dose calculations or setpoint determinations; and

   c. Documentation of the fact that the change has been reviewed and found acceptable by the POSRC.

2. Shall become effective upon review by the POSRC and approval of the Plant Superintendent.

6.17 MAJOR CHANGES TO RADIOACTIVE LIQUID, GASEOUS AND SOLID WASTE TREATMENT SYSTEMS

6.17.1 Licensee initiated major changes to the radioactive waste systems (liquid, gaseous and solid) shall be reported to the Commission in the Semiannual Radioactive Effluent Release Report for the period in which the modification to the waste system is completed. The discussion of each change shall contain:

a. A description of the equipment, components and processes involved.

b. Documentation of the fact that the change including the safety analysis was reviewed and found acceptable by the POSRC.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED AMENDMENT NOS. 108 AND 91

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 amendment dated June 28, 1985, 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 to: (1) reflect a clarification of requirements associated with the containment purge isolation valves in TS Table 3.3-3, "Engineered Safety Features Actuation System Instrumentation," and TS Table 3.6-1, "Containment Isolation Valves," (2) modify TS 3.9.4, "Containment Penetration," to allow the use of an alternate closure for the emergency personnel escape lock, (3) delete TS 6.13, "Environmental Qualifications," and (4) correct identified spelling errors and changes in terminology.

Additional changes to the TS that were requested in the June 28, 1985 application will be addressed in subsequent correspondence.

Discussion and Evaluation

BG&E has requested a change to TS Tables 3.3-3 and 3.6-1 to correct an inconsistency between operability requirements for the containment purge isolation valves and related requirements.

The containment purge isolation valves allow outside air to enter the containment and vent the containment atmosphere to the environment. At the present time, these valves are required to be isolated by the requirements of TS 3.6.1.7, "Containment Purge System," to prevent these valves from being opened, during Modes 1 through 4 (power operation through hot shutdown). Furthermore, the purge isolation valves are required to be operable, meaning capable of automatic closure to a leak-tight condition, by the requirements of TS 3.9.9. "Containment Purge Valve Isolation System," during core alterations or movement of irradiated fuel inside the containment (during Mode 6). A comparison of the requirements of TS 3.6.1.7 and TS 3.9.9 indicates that the containment purge isolation valves may be inoperable (open) in Modes 5 (cold shutdown) or in Mode 6 (refueling) when neither core alterations nor movement of irradiated

fuel inside containment is underway or any time in Mode 6 when the containment purge valves are closed. Conditions under which the containment isolation valves may be inoperable are consistent with conditions under which containment (leak-tight) integrity is not required per TS 3.6.1.1 and TS 3.9.4.

BG&E has identified two inconsistencies regarding requirements associated with the containment purge isolation valves. The first instance of inconsistency involves TS Table 3.6-1. In this case, the TS requires that the valve isolation response time for the containment purge isolation valves be applicable "... for Mode 5 and 6 during which time these valves may be opened." Since isolation response times should not be applicable when the valves are not required to be operable, BG&E has proposed to reword the applicability of the response time to be "... in Mode 6 when the valves are required to be operable and they are open." This proposed applicability wording is consistent with operability requirements of the containment purge isolation valves.

The second instance of inconsistency involves TS Table 3.3-3 which specifies operability requirements for devices for manual and automatic closure of the containment purge isolation valves. At the present time, the purge valve control switches for manual closure must be operable in Modes 5 and 6 and containment radiation-high area monitor (for automatic closure) must be operable in Mode 6. The licensee has proposed changing the operability requirements for these closure devices to "... in Mode 6 when the valves are required operable and they are open" which is consistent with the operability requirements for the containment purge isolation valves.

The proposed changes to TS Tables 3.6-1 and 3.3-3 would be consistent with analyses for which closure of the containment purge isolation valves is assumed. Operability of automatic and manual valve closure devices and closure response times would be applicable at all times when the containment purge isolation valves are required to be operable (these requirements only apply to the containment purge isolation valves). Accordingly, the proposed changes to TS Tables 3.3-3 and 3.6-1 are acceptable.

The licensee has proposed a change to TS 3.9.4b which would provide a footnote to allow use of a temporary closure for the containment emergency personnel escape lock during refueling activities. At the present time, TS 3.9.4 requires at least one door in each air lock to be closed during core alterations or movement of irradiated fuel inside the containment.

The personnel escape lock described in Section 5.1.2.1 of the Calvert Cliffs Final Safety Analysis Report (FSAR) is located at elevation 49'4" and is provided with outer and inner doors. During refueling operations, the licensee proposes to replace the inner personnel escape lock door with a temporary closure; the outer door would remain open at this time. This temporary closure would contain several penetrations to facilitate work inside containment, during core alterations or movement of irradiated fuel, when containment integrity is required. The licensee has indicated that the temporary closure and its

penetrations meet the applicable design requirements of the permanent door for use during reactor Mode 6 (refueling). Installation and leak testing of the temporary closure would be controlled by a plant procedure.

The Bases for TS 3/4.9.4 states the following with regard to containment closures such as the personnel escape lock during refueling: "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." Since no containment pressurization results from the design basis (fuel handling) event during Mode 6, containment closures need only be vapor-tight rather than capable of withstanding excess pressure. Since the temporary closure is fabricated to standards equivalent to the personnel escape lock for Mode 6, utilization and installation and testing will be in accordance with plant procedures, the temporary closure can be expected to perform in a manner equivalent to that of the personnel escape lock door during the design basis event in Mode 6. Moreover, since either a personnel escape lock door or the equivalent (temporary closure) will be in place during core alterations or movement of irradiated fuel inside containment, current, approved, analysis concerning fuel handling accidents is still applicable. Accordingly, the proposed change to TS 3.9.4b., to allow use of a temporary closure device, is acceptable.

The licensee has proposed to delete TS 6.13, "Environmental Qualifications." This TS provides schedule requirements for completion of activities relating to environmental qualification of electrical equipment important to safety that have already passed. Moreover, environmental qualification requirements, including schedules, are incorporated in 10 CFR 50.49, "Environmental qualification of electric equipment important to safety for nuclear power plants," and thus need not appear in the TS and, accordingly, TS 6.13 can be deleted.

Finally the licensee proposes to correct 14 spelling and one terminology error in the TS as detailed in the June 28, 1985 application. The correction of these spelling and terminology errors do not affect the associated TS requirements and 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. 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 4, 1985

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