December 9, 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. 109 and 92 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 April 26, 1985 as supplemented by your letter dated September 30, 1985.

The amendments change the Unit 1 and Unit 2 Technical Specifications (TS) to: (1) reflect a clarification of surveillance requirements of TS 4.6.1.6.2, "Containment Structural Integrity", concerning containment tendon end anchorages and adjacent concrete surfaces and a change to TS 4.6.1.6.3, "Liner Plate"; (2) reflect an increase in the required diesel generator test load specified in TS 4.8.1.1.2.c.2, "A.C.Sources"; (3) delete TS 3/4 3.3.8, "Radioactive Gaseous Effluent Monitoring Instrumentation" and incorporate these requirements in TS Tables 3.3-6 and 4.3-3, "Radiation Monitoring Instrumentation"; (4) provide simplification, additions and clarifications concerning the fire protection instrumentation in TS Table 3.3-11, "Fire Protection Instruments"; (5) revise limiting conditions and surveillance requirements for the hydrogen analyzers, TS 3/4.6.5, "Combustible Gas Control-Hydrogen Analyzers"; and (6) revise limiting conditions and surveillance requirements for the auxiliary feedwater system (TS 3/4.7.1.2).

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

8512270061 851209
PDR ADOCK 05000317
P PDR

Enclosures:
1. Amendment No. 109 to DPR-53
2. Amendment No. 92 to DPR-69
3. Safety Evaluation

cc w/enclosure:
See next page

		AD:CPS*	AD:C&SE*	
		LRubenstein	RBosnak	
		9/12/85	9/23/85	
ORB#3:DL*	ORB#3:DL*	ORB#3:DL*	OELD*	AD:OR:DL*
PMKreutzer	DJaffe	EButcher	CWoodhead	GCLainas
11/5/85	11/5/85	11/7/85	11/8/85	11/20/85

*See previous white for concurrences

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

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Sincerely,

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

AD:CPS*	AD:C&SE*
LRubenstein	RBosnak
9/12/85	9/23/85

No issuance until notice expires 12/4/85

ORB#3:DL	ORB#3:DL	ORB#3:DL	OELD	AD:OR:DL
PMKreutzer	DJaffe	EButcher		GCLainas
11/5/85	11/5/85	11/7/85	11/4/85	11/26/85

*See previous white for concurrences

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. 109
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 April 26, 1985 as supplemented by letter dated September 30, 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. 109, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

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

FOR THE NUCLEAR REGULATORY COMMISSION

A C Thadani

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

Attachment:
Changes to the Technical
Specifications

Date of Issuance: December 9, 1985

ATTACHMENT TO LICENSE AMENDMENT NO. 109

FACILITY 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
IV	IV
3/4 3-26	3/4 3-26
3/4 3-28	3/4 3-28
3/3 3-45	3/4 3-45
3/4 3-46	3/4 3-46
3/4 3-47	3/4 3-47
3/4 6-9	3/4 6-9
3/4 6-26	3/4 6-26
3/4 7-5	3/4 7-5
3/4 7-5b	3/4 7-5b
3/4 8-3	3/4 8-3
B 3/4 3-2	B 3/4 3-2
B 3/4 3-3	B 3/4 3-3
B 3/4 6-2	B 3/4 6-2

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

INDEX

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

3/4.3.3 MONITORING INSTRUMENTATION

RADIATION MONITORING INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.3.1 The radiation monitoring instrumentation channels shown in Table 3.3-6 shall be OPERABLE with their alarm/trip setpoints within the specified limits.

APPLICABILITY: As shown in Table 3.3-6.

ACTION:

a. With a radiation monitoring channel alarm/trip setpoint exceeding the value shown in Table 3.3-6, adjust the setpoint to within the limit within 4 hours or declare the channel inoperable.

b. With one or more radiation monitoring channels inoperable, take the ACTION shown in Table 3.3-6.

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

SURVEILLANCE REQUIREMENTS

4.3.3.1 Each radiation monitoring instrumentation channel shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL CALIBRATION and CHANNEL FUNCTIONAL TEST operations during the modes and at the frequencies shown in Table 4.3-3.

TABLE 3.3-6

RADIATION MONITORING INSTRUMENTATION

INSTRUMENT	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ALARM/TRIP SETPOINT	MEASUREMENT RANGE	ACTION
1. AREA MONITORS					
a. Containment					
i. Purge & Exhaust Isolation	3	6	$\leq$ 220 mr/hr	10^{-1} - 10^{4} mr/hr	16
b. Containment Area High Range	2	1, 2, 3, & 4	$\leq$ 10 R/hr	1 - 10^{8} R/hr	30
2. PROCESS MONITORS					
a. Containment					
i. Gaseous Activity					
a) RCS Leakage Detection	1	1, 2, 3, & 4	Not Applicable	1 - 10^{6} cpm	14
ii. Particulate Activity					
a) RCS Leakage Detection	1	1, 2, 3, & 4	Not Applicable	1 - 10^{6} cpm	14
b. Effluent Monitors					
i. Main Vent Wide Range					
a) Noble Gas	1	1, 2, 3, & 4	*	10^{-7} to 10^{+5} μCi/cc	30
b) Iodine Sampler	1	1, 2, 3, & 4	Not Applicable	Not Applicable	30
c) Particulate Sampler	1	1, 2, 3, & 4	Not Applicable	Not Applicable	30

*Alarm setpoint to be specified in a controlled document (e.g., setpoint control manual)

TABLE 3.3-6 (Continued)

TABLE NOTATION

ACTION 14 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, comply with the ACTION requirements of Specification 3.4.6.1.

ACTION 16 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, comply with the ACTION requirements of Specification 3.9.9.

ACTION 30 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, initiate the preplanned alternate method of monitoring the appropriate parameter(s), within 72 hours, and:

1) either restore the inoperable channel(s) to OPERABLE status within 7 days of the event, or

2) prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within 30 days following the event, outlining the action taken, the cause of the inoperability, and the plans and schedule for restoring the system to OPERABLE status.

TABLE 4.3-3

RADIATION MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES IN WHICH SURVEILLANCE REQUIRED
1. AREA MONITORS				
a. Containment				
i. Purge & Exhaust Isolation	S	R	M	6
b. Containment Area High Range	S	R	M	1, 2, 3, & 4
2. PROCESS MONITORS				
a. Containment				
i. Gaseous Activity a) RCS Leakage Detection	S	R	M	1, 2, 3, & 4
ii. Particulate Activity a) RCS Leakage Detection	S	R	M	1, 2, 3, & 4
b. Effluent Monitor				
i. Main Vent Wide Range a) Noble Gas	S	R	M	1, 2, 3, & 4
b) Iodine Sampler	M*	Not Applicable	Not Applicable	1, 2, 3, & 4
c) Particulate Sampler	M*	Not Applicable	Not Applicable	1, 2, 3, & 4

*The CHANNEL CHECK shall be accomplished by comparing samples independently drawn from the main vent.

TABLE 3.3-11

FIRE DETECTION INSTRUMENTS

UNIT 1

ROOM/AREA AUX BLDG	INSTRUMENT LOCATION	MINIMUM INSTRUMENTS OPERABLE* HEAT	FLAME	SMOKE
100/103/ 104/106	Corridors - Elev (-)10"-0"			5
110	Coolant Waste Rec & Mon. Tk Pp Rm			2
111	Waste Processing Control Rm		4	
112/114	Coolant Waste Rec Tank			1
115	Charging Pump Room			3
118/122	ECCS Pump Room			7
119/123	ECCS Pump Room			7
200/202	Corridors, &			
209/210	Corridors &			
212/219	Corridors			13
207/208	Waste Gas Equip Rm			3
216	Reactor Coolant Make-up Pumps			1
217	Boric Acid Tank & Pump Room			2
218	Volume Control Tank Room			1
220	Degasifier Pump Room			1
221/326	West Piping Penetration Room		2	3
222	Hot Instrument Shop			2
223	Hot Machine Shop			4
224	12 MSIV Hyd Area			10
225	Rad Exhaust Vent Equip Rm			4
226	Service Water Pump Rm		3	6
227/316	East, Piping Penetration Rm		3	5
228	Component Cooling Pump Rm			8
301/304/300	Battery Room & Corridor			3
306/1C	Cable Spreading Rm & Cable Chase**	2		10
308	N/S Corridor			6
315	Main Steam Piping Area			6
317	Switchgear Room, Elev 27'-0"**			6
318	Purge Air Supply Room			2
319/325	West Passage and Vestibule			6
320	Spent Fuel Heat Exchange Room			3
323	Passage 27' Valve Alley & Filter Rm			3
324	Letdown Heat Exchanger Rm			1
Elev. 27'-0"	Switchgear Vent Duct	1		
1A	Cable Chase 1A			1
1B	Cable Chase 1B			1
405	Control Room			6
410	N/S Corridor			4
417/418	Solid Waste Processing		2	3

TABLE 3.3-11 (Continued)

FIRE DETECTION INSTRUMENTS

UNIT 1

ROOM/AREA AUX BLDG	INSTRUMENT LOCATION	MINIMUM INSTRUMENTS OPERABLE* HEAT	FLAME	SMOKE
413/419/420	Cask and Equip Loading Area &			
424/425/426	Cask and Equip Loading Area		3	22
421	Diesel Generator No. (12)**	2		
422	Diesel Generator No. (11)**	2		
423	West Electrical Pen Rm			3
428	East Piping Area			7
429	East Electrical Pene Rm			3
430	Switchgear Room Elev 45'-0"**			8
439	Refueling Water Tank Pump Rm			2
441	Spent Resin Metering Tank Rm			1
Elev 45'-0"	Switchgear Vent Duct	1		
Elev 69'-0"	Control Room Vent Duct "A"			1
Elev 69'-0"	Cable Spreading Room Vent Duct			1
512	Control Room HVAC Equipment			4
586-590,	Radiation Chemistry Area,			
592,593	Radiation Chemistry Area,			
595-597,	Radiation Chemistry Area &			
521,523	Corridors			20
520	Spent Fuel Pool Area Vent Equip Rm			2
524	Main Plant Exhaust Equip Rm			8
525	Cntmt Access Area			3
529	Electrical Equip. Room			3
530/531/533	Spent Fuel Pool Area		5	17
536/537	Misc Waste Evaporator & Equip Rm			3
Elev 83'-0"	Cable Tunnel			4
603	Auxiliary Feedwater Pump Rm			2
Containment Bldgs				
U-1	RCP Bay East*	16		
U-1	RCP Bay West*	16		
U-1	East Electric Pen Area*	4***		
U-1	West Electric Pen Area*	4***		
Intake Structure	Elev 3'-0" Unit 1 Side			24

* Detection instruments located within the containment are not required to be OPERABLE during the performance of Type A Containment Leakage Rate Tests.

** Detectors which automatically actuate fire suppression systems.

*** Monitored by four protecto wires.

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In addition, determining that the average of the normalized lift-off forces for each sample population (hoop, vertical, dome) is equal to or greater than the required average prestress level; 536 kips for hoop tendons, 622 kips for vertical tendons, and 555 kips for dome tendons (reference Figures 4.6-1, -2, and -3). If the average is below the required average prestress force, it shall be considered as evidence of possible abnormal degradation of the containment structure.

b. Removing one wire from each of a dome, vertical and hoop tendon checked for lift off force, and determining over the entire length of the wire:

1. The extent of corrosion, cracks, or other damage. The presence of abnormal corrosion, cracks or other damage shall be considered evidence of possible abnormal degradation of the containment structure.

2. A minimum tensile strength value of 240 Ksi (guaranteed ultimate strength of the tendon material) for at least three wire samples (one from each end and one at mid-length) cut from each removed wire. Failure of any one of the wire samples to meet the minimum tensile strength test is evidence of possible abnormal degradation of the containment structure.

c. Perform a chemical analysis to detect changes in the chemical properties of the sheath filler grease. Any unusual changes in physical appearance or chemical properties that could adversely affect the ability of the filler grease to adhere to the tendon wires or otherwise inhibit corrosion shall be reported to the Commission pursuant to Specification 6.9.2 within the next 30 days.

4.6.1.6.2 End Anchorages and Adjacent Concrete Surfaces. The structural integrity of the end anchorages and adjacent concrete surfaces shall be demonstrated by determining through inspection of a representative sample of tendons (reference Specification 4.6.1.6.1) that no apparent changes have occurred in the visual appearance of the end anchorages or their adjacent concrete exterior surfaces. Also, inspections of the pre-selected concrete crack patterns adjacent to end anchorages shall be performed during the Type A containment leakage rate tests (reference Specification 4.6.1.2) while the containment is at its maximum test pressure.

4.6.1.6.3 Containment Surfaces. The exposed accessible interior and exterior surfaces of the containment, including the liner plate shall be visually inspected during the shutdown for each Type A containment leakage rate test (reference Specification 4.6.1.2). This inspection shall be performed prior to the Type A containment leakage rate test to uncover any evidence of structural deterioration which may affect either the containment structural integrity or leak tightness.

4.6.1.6.4 Reports. Any abnormal degradation of the containment structure detected during the above required tests and inspections shall be reported to the Commission pursuant to Specification 6.9.2 within the next 30 days. This report shall include a description of the tendon condition, the condition of the concrete (especially at tendon anchorages), the inspection procedures, the tolerances on cracking, and the corrective actions taken.

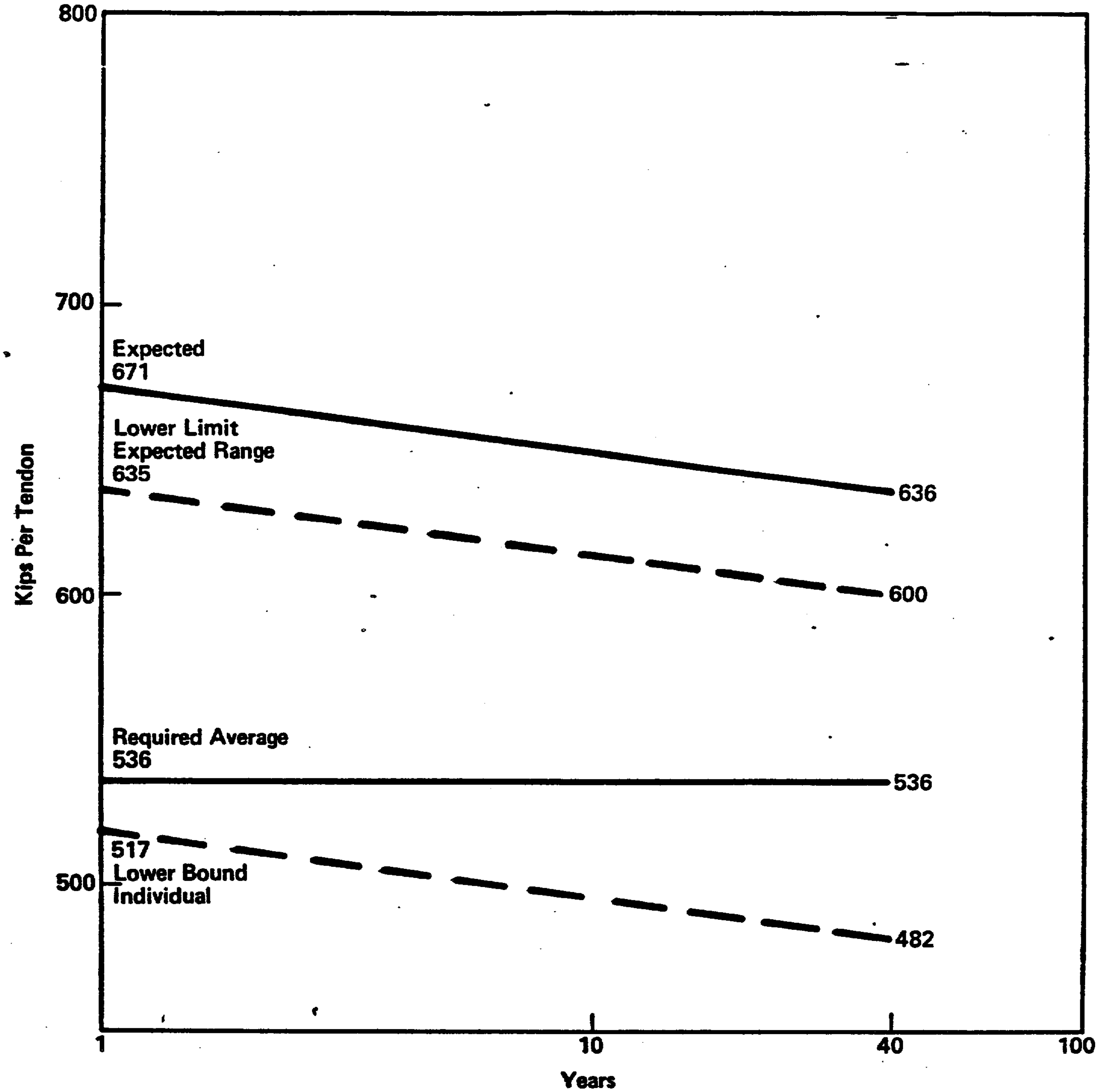

Figure 4.6-1 Normalized Prestress Hoop Tendons

TABLE 3.6-1 (Continued)

CONTAINMENT ISOLATION VALVES

PENETRATION NO.	ISOLATION CHANNEL	ISOLATION VALVE IDENTIFICATION NO.	FUNCTION	ISOLATION TIME (SECONDS)
61	NA NA NA NA	SFP-176 SFP-174 SFP-172 SFP-189	Refueling Pool Outlet	NA NA NA NA
62	SIAS A	PH-6579-MOV	Containment Heating Outlet	$\leq$13
64	NA	PH-376	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 for MODES 5 and 6 during which time these valves may be opened. 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.

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. 13 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. 13 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. 23 motor-driven pump is OPERABLE and valve 2-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)

ACTION: (Continued)

c. Whenever a subsystem(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 driven 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, and

2. Verifying that the auxiliary feedwater system is capable of providing a minimum of 200 gpm nominal flow to each flow leg.

SURVEILLANCE REQUIREMENTS (Continued)

a. At least once per 31 days on a STAGGERED TEST BASIS by:

1. Verifying the fuel level in the day fuel tank.

2. Verifying the fuel level in the fuel storage tank.

3. Verifying the fuel transfer pump can be started and transfers fuel from the storage system to the day tank.

4. Verifying the diesel starts from ambient condition and accelerates to at least 900 rpm in ≤ 10 seconds.

5. Verifying the generator is synchronized, loaded to ≥ 1250 kw, and operates for ≥ 60 minutes.

6. Verifying the diesel generator is aligned to provide standby power to the associated emergency busses.

7. Verifying that the automatic load sequence timer is OPERABLE with the interval between each load block within ± 10% of its design interval.

b. At least once per 92 days by verifying that a sample of diesel fuel from the fuel storage tank is within the acceptable limits specified in Table 1 of ASTM D975-68 when checked for viscosity, water and sediment.

c. At least once per 18 months by:

1. Subjecting the diesel to an inspection in accordance with procedures prepared in conjunction with its manufacturer's recommendations for this class of standby service.

2. Verifying the generator capability to reject a load of ≥ 500 hp without tripping.

3. Simulating a loss of offsite power in conjunction with a safety injection actuation test signal, and:

a) Verifying de-energization of the emergency busses and load shedding from the emergency busses.

b) Verifying the diesel starts from ambient condition on the auto-start signal, energizes the emergency busses with permanently connected loads, energizes the auto-connected emergency loads through the load sequencer and operates for $\geq$ 5 minutes while its generator is loaded with the emergency loads.

c) Verifying that all diesel generator trips, except engine overspeed, crankcase pressure high, lube oil pressure low, generator ground overcurrent, and generator differential, are automatically bypassed on a Safety Injection Actuation Signal.

4. Verifying the diesel generator operates for $\geq$ 60 minutes while loaded to $\geq$ 2500 kw.

5. Verifying that the auto-connected loads to each diesel generator do not exceed the 2000 hour rating of 2700 kw.

3/4.3.1 and 3/4.3.2 PROTECTIVE AND ENGINEERED SAFETY FEATURES (ESF) INSTRUMENTATION

The OPERABILITY of the protective and ESF instrumentation systems and bypasses ensure that 1) the associated ESF action and/or reactor trip will be initiated when the parameter monitored by each channel or combination therof exceeds its setpoint, 2) the specified coincidence logic is maintained, 3) sufficient redundancy is maintained to permit a channel to be out of service for testing or maintenance, and 4) sufficient system functional capability is available for protective and ESF purposes from diverse parameters.

The OPERABILITY of these systems is required to provide the overall reliability, redundance and diversity assumed available in the facility design for the protection and mitigation of accident and transient conditions. The integrated operation of each of these systems is consistent with the assumptions used in the accident analyses.

The surveillance requirements specified for these systems ensure that the overall system functional capability is maintained comparable to the original design standards. The periodic surveillance tests performed at the minimum frequencies are sufficient to demonstrate this capability.

The measurement of response time at the specified frequencies provides assurance that the protective and ESF action function associated with each channel is completed within the time limit assumed in the accident analyses. No credit was taken in the analyses for those channels with response times indicated as not applicable.

Response time may be demonstrated by any series of sequential, overlapping or total channel test measurements provided that such tests demonstrate the total channel response time as defined. Sensor response time verification may be demonstrated by either 1) in place, onsite or offsite test measurements or 2) utilizing replacement sensors with certified response times.

3/4.3.3 MONITORING INSTRUMENTATION

3/4.3.3.1 RADIATION MONITORING INSTRUMENTATION

The OPERABILITY of the radiation monitoring channels ensures that 1) the radiation levels are continually measured in the areas served by the individual channels and 2) the alarm or automatic action is initiated when the radiation level trip setpoint is exceeded.

The Iodine and Particulate samplers were installed to meet the requirements of NUREG-0737 Item II.F.1. The samplers' operation was not assumed in any accident analysis.

3/4.3.3.2 INCORE DETECTORS

The OPERABILITY of the incore detectors with the specified minimum complement of equipment ensures that the measurements obtained from use of this system accurately represent the spatial neutron flux distribution of the reactor core.

3/4.3.3.3. SEISMIC INSTRUMENTATION

The OPERABILITY of the seismic instrumentation ensures that sufficient capability is available to promptly determine the magnitude of a seismic event and evaluate the response of those features important to safety. This capability is required to permit comparison of the measured response to that used in the design basis for the facility and is consistent with the recommendations of Regulatory Guide 1.12, "Instrumentation for Earthquakes," April 1974.

3/4.3.3.4 METEOROLOGICAL INSTRUMENTATION

The OPERABILITY of the meteorological instrumentation ensures that sufficient meteorological data is available for estimating potential radiation doses to the public as a result of routine or accidental release of radioactive materials to the atmosphere. This capability is required to evaluate the need for initiating protective measures to protect the health and safety of the public and is consistent with the recommendations of Regulatory Guide 1.23 "Onsite Meteorological Programs," February 1972.

3/4.3.3.5 REMOTE SHUTDOWN INSTRUMENTATION

The OPERABILITY of the remote shutdown instrumentation ensures that sufficient capability is available to permit shutdown and maintenance of HOT STANDBY of the facility from locations outside of the control room. This capability is required in the event control room habitability is lost and is consistent with General Design Criteria 19 of 10 CFR 50.

3/4.3.3.6 POST-ACCIDENT INSTRUMENTATION

The OPERABILITY of the post-accident instrumentation ensures that sufficient information is available on selected plant parameters to monitor and assess these variables following an accident. This capability is consistent with the recommendations of Regulatory Guide 1.97, "Instrumentation for Light-Water-Cooled Nuclear Plants to Assess Plant Conditions During and Following an Accident," December 1975, and NUREG-0578, "TMI-2 Lessons Learned Task Force Status Report and Short-Term Recommendations."

3/4.3.3.7 FIRE DETECTION INSTRUMENTATION

OPERABILITY of the fire detection instrumentation ensures that adequate warning capability is available for the prompt detection of fires. This capability is required in order to detect and locate fires in their early stages. Prompt detection of fires will reduce the potential for damage to safety related equipment and is an integral element in the overall facility fire protection program.

In the event that a portion of the fire detection instrumentation is inoperable, the establishment of frequent fire patrols in the affected areas is required to provide detection capability until the inoperable instrumentation is restored to operability.

3/4.3.3.9 RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION

The radioactive gaseous effluent instrumentation is provided to monitor and control, as applicable, the releases of radioactive materials in gaseous effluents during actual or potential releases of gaseous effluents. The alarm/trip setpoints for these instruments shall be calculated and adjusted in accordance with the methodology and parameters in the ODCM to ensure that the alarm/trip will occur prior to exceeding the limits of Specification 3.11.2.1.a based on average annual X/Q. The OPERABILITY and use of this instrumentation is consistent with the requirements of General Design Criteria 60, 63, and 64 of Appendix A to 10 CFR Part 50.

3/4.3.3.10 RADIOACTIVE LIQUID EFFLUENT MONITORING INSTRUMENTATION

The radioactive liquid effluent instrumentation is provided to monitor and control, as applicable, the releases of radioactive materials in liquid effluents during actual or potential releases of liquid effluents. The alarm/trip setpoints for these instruments shall be calculated and adjusted in accordance with the methodology and parameters in the ODCM to ensure that the alarm/trip will occur prior to exceeding the limits of 10 CFR Part 20. The OPERABILITY and use of this instrumentation is consistent with the requirements of General Design Criteria 60, 63 and 64 of Appendix A to 10 CFR Part 50.

3/4.6.1 PRIMARY CONTAINMENT

3/4.6.1.1 CONTAINMENT INTEGRITY

Primary CONTAINMENT INTEGRITY ensures that the release of radioactive materials from the containment atmosphere will be restricted to those leakage paths and associated leak rates assumed in the accident analyses. This restriction, in conjunction with the leakage rate limitation, will limit the site boundary radiation doses to within the limits of 10 CFR 100 during accident conditions.

3/4.6.1.2 CONTAINMENT LEAKAGE

The limitations on containment leakage rates ensure that the total containment leakage volume will not exceed the value assumed in the accident analyses at the peak accident pressure, P_a. As an added conservatism, the measured overall integrated leakage rate is further limited to $\leq 0.75\ L_a$ or $\leq 0.75\ L_t$ (as applicable) during performance of the periodic tests to account for possible degradation of the containment leakage barriers between leakage tests.

The surveillance testing for measuring leakage rates are consistent with the requirements of Appendix "J" of 10 CFR 50.

3/4.6.1.3 CONTAINMENT AIR LOCKS

The limitations on closure and leak rate for the containment air locks are required to meet the restrictions on CONTAINMENT INTEGRITY and containment leak rate. Surveillance testing of the air lock seals provides assurance that the overall air lock leakage will not become excessive due to seal damage during the intervals between air lock leakage tests.

3/4.6.1.4 INTERNAL PRESSURE

The limitations on containment internal pressure ensure that 1) the containment structure is prevented from exceeding its design negative pressure differential with respect to the outside atmosphere of 3.0 psig and 2) the containment peak pressure does not exceed the design pressure of 50 psig during LOCA conditions.

The maximum peak pressure expected to be obtained from a LOCA event is 47.6 psig. The limit of 1.8 psig for initial positive containment pressure will limit the total pressure to 49.4 psig which is less than the design pressure and is consistent with the accident analyses.

3/4.6.1.5 AIR TEMPERATURE

The limitation on containment average air temperature ensures that the containment peak air temperature does not exceed the design temperature of 276°F during LOCA conditions. The containment temperature limit is consistent with the accident analyses.

3/4.6.1.6 CONTAINMENT STRUCTURAL INTEGRITY

This limitation ensures that the structural integrity of the containment will be maintained comparable to the original design standards for the life of the facility. Structural integrity is required to ensure that the containment will withstand the maximum pressure of 47.6 psig in the event of a LOCA. The measurement of containment tendon lift off force, the visual and metallurgical examination of tendons, anchorages and liner and the Type A leakage tests are sufficient to demonstrate this capability.

The surveillance requirements for demonstrating the containment's structural integrity are consistent with the intent of the recommendations of Regulatory Guide 1.35 "Inservice Surveillance of Ungrouted Tendons in Prestressed Concrete Containment Structures", January 1976.

The end anchorage concrete exterior surfaces are checked visually for indications of abnormal material behavior during tendon surveillance. Inspections of pre-selected concrete crack patterns are performed during the Type A containment leakage rate tests, consistent with the Structural Integrity Test.

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. 92
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 April 26, 1985 as supplemented by letter dated September 30, 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. 92, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

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

FOR THE NUCLEAR REGULATORY COMMISSION

A C Thadani

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

Attachment:
Changes to the Technical
Specifications

Date of Issuance: December 9, 1985

ATTACHMENT TO LICENSE AMENDMENT NO. 92

FACILITY 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
IV	IV
3/4 3-26	3/4 3-26
3/4 3-28	3/4 3-28
3/3 3-45	3/4 3-45
3/4 3-46	3/4 3-46
3/4 3-47	3/4 3-47
3/4 6-8	3/4 6-8
3/4 6-9	3/4 6-9
3/4 6-26	3/4 6-26
3/4 7-5a	3/4 7-5a
3/4 8-3	3/4 8-3
B 3/4 3-2	B 3/4 3-2
B 3/4 3-3	B 3/4 3-3
B 3/4 6-2	B 3/4 6-2

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LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

INDEX

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

3/4.3.3 MONITORING INSTRUMENTATION

RADIATION MONITORING INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.3.1 The radiation monitoring instrumentation channels shown in Table 3.3-6 shall be OPERABLE with their alarm/trip setpoints within the specified limits.

APPLICABILITY: As shown in Table 3.3-6.

ACTION:

a. With a radiation monitoring channel alarm/trip setpoint exceeding the value shown in Table 3.3-6, adjust the setpoint to within the limit within 4 hours or declare the channel inoperable.

b. With one or more radiation monitoring channels inoperable, take the ACTION shown in Table 3.3-6.

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

SURVEILLANCE REQUIREMENTS

4.3.3.1 Each radiation monitoring instrumentation channel shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL CALIBRATION and CHANNEL FUNCTIONAL TEST operations during the modes and at the frequencies shown in Table 4.3-3.

TABLE 3.3-6

RADIATION MONITORING INSTRUMENTATION

INSTRUMENT	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ALARM/TRIP SETPOINT	MEASUREMENT RANGE	ACTION
1. AREA MONITORS					
a. Containment					
i. Purge & Exhaust Isolation	3	6	≤ 220 mr/hr	10^{-4} - 10^{4} mr/hr	16
b. Containment Area High Range	2	1, 2, 3 & 4	≤ 10 R/hr	1 - 10^{8} R/hr	30
2. PROCESS MONITORS					
a. Containment					
i. Gaseous Activity					
a) RCS Leakage Detection	1	1, 2, 3 & 4	Not Applicable	10^{1} - 10^{6} cpm	14
ii. Particulate Activity					
a) RCS Leakage Detection	1	1, 2, 3 & 4	Not Applicable	10^{1} - 10^{6} cpm	14
b. Effluent Monitors					
i. Main Vent Wide Range					
a) Noble Gas	1	1, 2, 3 & 4	*	10^{-7} to 10^{5} µCi/cc	30
b) Iodine Sampler	1	1, 2, 3 & 4	Not Applicable	Not Applicable	30
c) Particulate Sampler	1	1, 2, 3 & 4	Not Applicable	Not Applicable	30

*Alarm setpoint to be specified in a controlled document (e.g., setpoint control manual).

TABLE 3.3-6 (Continued)

TABLE NOTATION

ACTION 14 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, comply with the ACTION requirements of Specification 3.4.6.1.

ACTION 16 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, comply with the ACTION requirements of Specification 3.9.9.

ACTION 30 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, initiate the preplanned alternate method of monitoring the appropirate parameter(s), within 72 hours, and:

1) either restore the inoperable channel(s) to OPERABLE status within 7 days of the event, or

2) prepare and sumbit a Special Report to the Commission pursuant to Specification 6.9.2 within 30 days following the event, outlining the action taken, the cause of the inoperability and the plans and schedule for restoring the system to OPERABLE status.

TABLE 4.3-3

RADIATION MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES IN WHICH SURVEILLANCE REQUIRED
1. AREA MONITORS				
a. Containment				
i. Purge & Exhaust Isolation	S	R	M	6
b. Containment Area High Range	S	R	M	1, 2, 3 & 4
2. PROCESS MONITORS				
a. Containment				
i. Gaseous Activity				
a) RCS Leakage Detection	S	R	M	1, 2, 3 & 4
ii. Particulate Activity				
a) RCS Leakage Detection	S	R	M	1, 2, 3 & 4
b. Effluent Monitors				
i. Main Vent Wide Range				
a) Noble Gas	S	R	M	1, 2, 3 & 4
b) Iodine Sampler	M*	Not Applicable	Not Applicable	1, 2, 3 & 4
c) Particulate Sampler	M*	Not Applicable	Not Applicable	1, 2, 3 & 4

*The CHANNEL CHECK shall be accomplished by comparing samples independently drawn from the main vent.

TABLE 3.3-11

FIRE DETECTION INSTRUMENTS

UNIT 2

ROOM/AREA AUX BLDG	INSTRUMENT LOCATION	MINIMUM INSTRUMENTS OPERABLE* HEAT	FLAME	SMOKE
101/120	ECCS Pump Room			7
102/121	ECCS Pump Room			7
105	Charging Pump Room			3
106	Misc Waste Monitor Tank			1
107/109	Coolant Waste Monitor Tank		4	
108	Pump Room-Elev (-)10'-0"			1
201	Component Cooling Pump Rm			9
203	East Piping Area			10
204	Rad Exhaust Vent, Equip Rm			4
205	Service Water Pump Rm		3	6
206/310	East Piping Pen Rm		3	5
211/321	West Piping Pen Rm		2	3
213	Degasifier Pump Rm			1
214	Volume Control Tank Rm			1
215	Boric Acid Tank & Pump Rm			2
216A	Reactor Coolant Make-up Pumps			2
302/2C	U2 Cable Spreading Rm & Cable Chase**	2		10
305/307/303	U2 Battery Rm & Corridor			3
309	Main Steam Piping Area			6
311	Switchgear Rm, Elev 27'-0"			6
312	Purge Air Supply Rm			2
322	Letdown Heat Exchanger Rm			1
Elev. 27'-0"	Switchgear Vent Duct	1		
2A	Cable Chase 2A			1
2B	Cable Chase 2B			1
407	Switchgear Rm, Elev 45'-0"**			8
408	East Piping Area			7
409	East Electrical Pen Rm			3
414	West Electrical Pen Rm			3
416	Diesel Generator No. (21)**	2		
440	Refueling Water Tank Pump Rm			2
Elev 45'-0"	Switchgear Vent Duct	1		
526	Main Plant Exhaust Equip Rm			8
527	Containment Access			3
532	Electrical Equip Rm			3
Elev. 69'-0"	Cable Spreading Room Vent Duct			1
Elev. 83'-0"	Cable Tunnel			4
605	Auxiliary Feedwater Pump Rm			2

TABLE 3.3-11 (Continued)

FIRE DETECTION INSTRUMENTS

UNIT 2

ROOM/AREA AUX BLDG	INSTRUMENT LOCATION	MINIMUM INSTRUMENTS OPERABLE* HEAT	FLAME	SMOKE
Containment Bldgs.				
UNIT 2	RCP Bay East*	16		
UNIT 2	RCP Bay West*	16		
UNIT 2	East Electric Pen Area*	+		
UNIT 2	West Electric Pen Area*	+		
Intake Structure Elev 3'-0" Unit 2 Side				24

*Detection instruments located within the containment are not required to be OPERABLE during the performance of Type A Containment Leakage Rate Tests.
**Detectors which automatically actuate fire suppression systems.
+Monitored by four protecto wires.

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CONTAINMENT SYSTEMS

AIR TEMPERATURE

LIMITING CONDITION FOR OPERATION

3.6.1.5 Primary containment average air temperature shall not exceed 120°F.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With the containment average air temperature > 120°F, reduce the average air temperature to within the limit within 8 hours, or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.1.5 The primary containment average air temperature shall be the arithmetical average of the temperatures at the following locations and shall be determined at least once per 24 hours:

Location

a. Containment Dome

b. Containment Reactor Cavity

CONTAINMENT STRUCTURAL INTEGRITY

LIMITING CONDITION FOR OPERATION

3.6.1.6 The structural integrity of the containment shall be maintained at a level consistent with the acceptance criteria in Specification 4.6.1.6.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With the structural integrity of the containment not conforming to the above requirements, restore the structural integrity to within the limits within 24 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.1.6.1 Containment Tendons. The containment tendons' structural integrity shall be demonstrated at the end of one, three and five years following the initial containment structural integrity test and at five year intervals thereafter. The tendons' structural integrity shall be demonstrated by a visual examination (to the extent practical and without dismantling load bearing components of the anchorage) of a representative sample of at least 21 tendons (6 dome, 5 vertical, and 10 hoop) and verifying no abnormal degradation. Unless there is evidence of abnormal degradation of the containment structure during the first three tests of the tendons, the number of tendons examined during subsequent tests may be reduced to a representative sample of at least 9 tendons (3 dome, 3 vertical and 3 hoop).

4.6.1.6.2 End Anchorages and Adjacent Concrete Surfaces. The structural integrity of the end anchorages and adjacent concrete surfaces shall be demonstrated by determining through inspection of a representative sample of tendons (reference Specification 4.6.1.6.1) that no apparent changes have occurred in the visual appearance of the end anchorages or their adjacent concrete exterior surfaces. Also, inspections of the pre-selected concrete crack patterns adjacent to end anchorages shall be performed during the Type A containment leakage rate tests (reference Specification 4.6.1.2) while the containment is at its maximum test pressure.

SURVEILLANCE REQUIREMENTS (Continued)

4.6.1.6.3 Containment Surfaces. The exposed accessible interior and exterior surfaces of the containment, including the liner plate shall be visually inspected during the shutdown for each Type A containment leakage rate test (reference Specification 4.6.1.2). This inspection shall be performed prior to the Type A containment leakage rate test to uncover any evidence of structural deterioration which may affect either the containment structural integrity or leak tightness.

4.6.1.6.4 Reports. Any abnormal degradation of the containment structure detected during the above required tests and inspections shall be reported to the Commission pursuant to Specification 6.9.2 within the next 30 days. This report shall include a description of the tendon condition, the condition of the concrete (especially at tendon anchorages), the inspection procedure, the tolerances on cracking, and the corrective actions taken.

CONTAINMENT SYSTEMS

CONTAINMENT PURGE SYSTEM

LIMITING CONDITION FOR OPERATION

3.6.1.7 The containment purge supply and exhaust isolation valves shall be closed by isolating air to the air operator and maintaining the solenoid air supply valve deenergized.

APPLICABILITY: MODES 1, 2, 3 and 4

ACTION:

a. With one containment purge supply and/or one exhaust isolation valve open, close the open valve(s) within one hour or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With one containment purge supply and/or one exhaust isolation valve inoperable due to high leakage, repair the valve(s) within 24 hours or be in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.1.7 The 48-inch containment purge supply and exhaust isolation valves shall be determined closed at least once per 31 days, by verifying that power to the solenoid valve is removed.

TABLE 3.6-1 (Continued)

CONTAINMENT ISOLATION VALVES

PENETRATION NO.	ISOLATION CHANNEL	ISOLATION VALVE IDENTIFICATION NO.	FUNCTION	ISOLATION TIME (SECONDS)
61	NA NA NA NA	SFP-184 SFP-182 SFP-180 SFP-186	Refueling Pool Outlet	NA NA NA 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.

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 subsystem(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 driven 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:

a. 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 200 gpm nominal flow to each flow leg.

SURVEILLANCE REQUIREMENTS (Continued)

a. At least once per 31 days on a STAGGERED TEST BASIS by:

   1. Verifying the fuel level in the day fuel tank.

   2. Verifying the fuel level in the fuel storage tank.

   3. Verifying the fuel transfer pump can be started and transfers fuel from the storage system to the day tank.

   4. Verifying the diesel starts from ambient condition and accelerates to at least 900 rpm in $\leq$ 10 seconds.

   5. Verifying the generator is synchronized, loaded to $\geq$ 1250 kw, and operates for $\geq$ 60 minutes.

   6. Verifying the diesel generator is aligned to provide standby power to the associated emergency busses.

   7. Verifying that the automatic load sequence timer is OPERABLE with the interval between each load block within $\pm$ 10% of its design interval.

b. At least once per 92 days by verifying that a sample of diesel fuel from the fuel storage tank is within the acceptable limits specified in Table 1 of ASTM D975-68 when checked for viscosity, water and sediment.

c. At least once per 18 months by:

   1. Subjecting the diesel to an inspection in accordance with procedures prepared in conjunction with its manufacturer's recommendations for this class of standby service.

   2. Verifying the generator capability to reject a load of $\geq$ 500 hp without tripping.

   3. Simulating a loss of offsite power in conjunction with a safety injection actuation test signal, and:

      1. Verifying de-energization of the emergency busses and load shedding from the emergency busses.

SURVEILLANCE REQUIREMENTS (Continued)

b) Verifying the diesel starts from ambient condition on the auto-start signal, energizes the emergency busses with permanently connected loads, energizes the auto-connected emergency loads through the load sequencer and operates for $\geq$ 5 minutes while its generator is loaded with the emergency loads.

c) Verifying that all diesel generator trips, except engine overspeed, crankcase pressure high, lube oil pressure low, generator ground overcurrent, and generator differential, are automatically bypassed on a Safety Injection Actuation Signal.

4. Verifying the diesel generator operates for $\geq$ 60 minutes while loaded to $\geq$ 2500 kw.

5. Verifying that the auto-connected loads to each diesel generator do not exceed the 2000 hour rating of 2700 kw.

3/4.3.1 and 3/4.3.2 PROTECTIVE AND ENGINEERED SAFETY FEATURES (ESF) INSTRUMENTATION

The OPERABILITY of the protective and ESF instrumentation systems and bypasses ensure that 1) the associated ESF action and/or reactor trip will be initiated when the parameter monitored by each channel or combination therof exceeds its setpoint, 2) the specified coincidence logic is maintained, 3) sufficient redundancy is maintained to permit a channel to be out of service for testing or maintenance, and 4) sufficient system functional capability is available for protective and ESF purposes from diverse parameters.

The OPERABILITY of these systems is required to provide the overall reliability, redundance and diversity assumed available in the facility design for the protection and mitigation of accident and transient conditions. The integrated operation of each of these systems is consistent with the assumptions used in the accident analyses.

The surveillance requirements specified for these systems ensure that the overall system functional capability is maintained comparable to the original design standards. The periodic surveillance tests performed at the minimum frequencies are sufficient to demonstrate this capability.

The measurement of response time at the specified frequencies provides assurance that the protective and ESF action function associated with each channel is completed within the time limit assumed in the accident analyses. No credit was taken in the analyses for those channels with response times indicated as not applicable.

Response time may be demonstrated by any series of sequential, overlapping or total channel test measurements provided that such tests demonstrate the total channel response time as defined. Sensor response time verification may be demonstrated by either 1) in place, onsite or offsite test measurements or 2) utilizing replacement sensors with certified response times.

3/4.3.3 MONITORING INSTRUMENTATION

3/4.3.3.1 RADIATION MONITORING INSTRUMENTATION

The OPERABILITY of the radiation monitoring channels ensures that 1) the radiation levels are continually measured in the areas served by the individual channels and 2) the alarm or automatic action is initiated when the radiation level trip setpoint is exceeded.

The iodine and particulate samplers were installed to meet the requirements of NUREG-0737 Item II.F.1. The samplers' operation was not assumed in any accident analysis.

3/4.3.3.2 INCORE DETECTORS

The OPERABILITY of the incore detectors with the specified minimum complement of equipment ensures that the measurements obtained from use of this system accurately represent the spatial neutron flux distribution of the reactor core.

3/4.3.3.3 SEISMIC INSTRUMENTATION

The OPERABILITY of the seismic instrumentation ensures that sufficient capability is available to promptly determine the magnitude of a seismic event and evaluate the response of those features important to safety. This capability is required to permit comparison of the measured response to that used in the design basis for the facility and is consistent with the recommendations of Regulatory Guide 1.12, "Instrumentation for Earthquakes," April 1974.

3/4.3.3.4 METEOROLOGICAL INSTRUMENTATION

The OPERABILITY of the meteorological instrumentation ensures that sufficient meteorological data is available for estimating potential radiation doses to the public as a result of routine or accidental release of radioactive materials to the atmosphere. This capability is required to evaluate the need for initiating protective measures to protect the health and safety of the public and is consistent with the recommendations of Regulatory Guide 1.23, Rev. 1 (Proposed), "Meteorological Programs in Support of Nuclear Power Plants," September 1980.

3/4.3.3.5 REMOTE SHUTDOWN INSTRUMENTATION

The OPERABILITY of the remote shutdown instrumentation ensures that sufficient capability is available to permit shutdown and maintenance of HOT STANDBY of the facility from locations outside of the control room. This capability is required in the event control room habitability is lost and is consistent with General Design Criteria 19 of 10 CFR 50.

3/4.3.3.6 POST-ACCIDENT INSTRUMENTATION

The OPERABILITY of the post-accident instrumentation ensures that sufficient information is available on selected plant parameters to monitor and assess these variables following an accident. This capability is consistent with the recommendations of Regulatory Guide 1.97, "Instrumentation for Light-Water-Cooled Nuclear Plants to Assess Plant Conditions During and Following an Accident," December 1975, and NUREG-0578, "TMI-2 Lessons Learned Task Force Status Report and Short-Term Recommendations."

3/4.3.3.7 FIRE DETECTION INSTRUMENTATION

OPERABILITY of the fire detection instrumentation ensures that adequate warning capability is available for the prompt detection of fires. This capability is required in order to detect and locate fires in their early stages. Prompt detection of fires will reduce the potential for damage to safety related equipment and is an integral element in the overall facility fire protection program.

In the event that a portion of the fire detection instrumentation is inoperable, the establishment of frequent fire patrols in the affected areas is required to provide detection capability until the inoperable instrumentation is restored to operability.

3/4.3.3.9 RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION

The radioactive gaseous effluent instrumentation is provided to monitor and control, as applicable, the releases of radioactive materials in gaseous effluents during actual or potential releases of gaseous effluents. The alarm/trip setpoints for these instruments shall be calculated and adjusted in accordance with the methodology and parameters in the ODCM to ensure that the alarm/trip will occur prior to exceeding the limits of Specification 3.11.2.1.a based on average annual X/Q. The OPERABILITY and use of this instrumentation is consistent with the requirements of General Design Criteria 60, 63, and 64 of Appendix A to 10 CFR Part 50.

3/4.3.3.10 RADIOACTIVE LIQUID EFFLUENT MONITORING INSTRUMENTATION

The radioactive liquid effluent instrumentation is provided to monitor and control, as applicable, the releases of radioactive materials in liquid effluents during actual or potential releases of liquid effluents. The alarm/trip setpoints for these instruments shall be calculated and adjusted in accordance with the methodology and parameters in the ODCM to ensure that the alarm/trip will occur prior to exceeding the limits of 10 CFR Part 20. The OPERABILITY and use of this instrumentation is consistent with the requirements of General Design Criteria 60, 63 and 64 of Appendix A to 10 CFR Part 50.

3/4.6.1 PRIMARY CONTAINMENT

3/4.6.1.1 CONTAINMENT INTEGRITY

Primary CONTAINMENT INTEGRITY ensures that the release of radioactive materials from the containment atmosphere will be restricted to those leakage paths and associated leak rates assumed in the accident analyses. This restriction, in conjunction with the leakage rate limitation, will limit the site boundary radiation doses to within the limits of 10 CFR 100 during accident conditions.

3/4.6.1.2 CONTAINMENT LEAKAGE

The limitations on containment leakage rates ensure that the total containment leakage volume will not exceed the value assumed in the accident analyses at the peak accident pressure, P_a. As an added conservatism, the measured overall integrated leakage rate is further limited to $\leq 0.75\ L_a$ or $\leq 0.75\ L_t$ (as applicable) during performance of the periodic tests to account for possible degradation of the containment leakage barriers between leakage tests.

The surveillance testing for measuring leakage rates are consistent with the requirements of Appendix "J" of 10 CFR 50.

3/4.6.1.3 CONTAINMENT AIR LOCKS

The limitations on closure and leak rate for the containment air locks are required to meet the restrictions on CONTAINMENT INTEGRITY and containment leak rate. Surveillance testing of the air lock seals provides assurance that the overall air lock leakage will not become excessive due to seal damage during the intervals between air lock leakage tests.

3/4.6.1.4 INTERNAL PRESSURE

The limitations on containment internal pressure ensure that 1) the containment structure is prevented from exceeding its design negative pressure differential with respect to the outside atmosphere of 3.0 psig and 2) the containment peak pressure does not exceed the design pressure of 50 psig during LOCA conditions.

The maximum peak pressure expected to be obtained from a LOCA event is 47.6 psig. The limit of 1.8 psig for initial positive containment pressure will limit the total pressure to 49.4 psig which is less than the design pressure and is consistent with the accident analyses.

3/4.6.1.5 AIR TEMPERATURE

The limitation on containment average air temperature ensures that the containment peak air temperature does not exceed the design temperature of 276°F during LOCA conditions. The containment temperature limit is consistent with the accident analyses.

3/4.6.1.6 CONTAINMENT STRUCTURAL INTEGRITY

This limitation ensures that the structural integrity of the containment will be maintained comparable to the original design standards for the life of the facility. Structural integrity is required to ensure that the containment will withstand the maximum pressure of 47.6 psig in the event of a LOCA. The measurement of containment tendon lift off force, the visual and metallurgical examination of tendons, anchorages and liner and the Type A leakage tests are sufficient to demonstrate this capability.

The surveillance requirements for demonstating the containment's structural integrity are consistent with the intent of the recommendations of Regulatory Guide 1.35 "Inservice Surveillance of Ungrouted Tendons in Prestressed Concrete Containment Structures," January 1976.

The end anchorage concrete exterior surfaces are checked visually for indications of abnormal material behavior during tendon surveillance. Inspections of pre-selected concrete crack patterns are performed during the Type A containment leakage rate tests, consistent with the Structural Integrity Test.

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NOS. 109 AND 92

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 April 26, 1985 as supplemented by letter dated September 30, 1985, Baltimore Gas & Electric Company (BG&E/licensee) requested changes to the Technical Specifications (TS) for Calvert Cliffs Units 1 and 2. The proposed TS changes would: (1) reflect a clarification of surveillance requirements of TS 4.6.1.6.2, "Containment Structural Integrity", concerning containment tendon end anchorages and adjacent concrete surfaces and a change to TS 4.6.1.6.3, "Liner Plate"; (2) reflect an increase in the required diesel generator test load specified in TS 4.8.1.1.2.c.2, "A.C.Sources"; (3) delete TS 3/4 3.3.8, "Radioactive Gaseous Effluent Monitoring Instrumentation" and incorporate these requirements in TS Tables 3.3-6 and 4.3-3, "Radiation Monitoring Instrumentation"; (4) provide simplification, additions and clarifications concerning the fire protection instrumentation in TS Table 3.3-11, "Fire Detection Instruments"; (5) revise limiting conditions and surveillance requirements for the hydrogen analyzers, TS 3/4.6.5, "Combustible Gas Control-Hydrogen Analyzers"; and (6) revise limiting conditions and surveillance requirements for the auxiliary feedwater system (TS 3/4.7.1.2).

Additional TS changes requested in the April 26, 1985 application will be addressed as part of a future license amendment.

Discussion and Evaluation

The licensee has requested a change to TS 4.6.1.6.2 in order to provide clarification regarding inspection of containment tendon end anchorages and adjacent concrete surfaces. The wording of TS 4.6.1.6.2 would seem to indicate that all end anchorages and adjacent concrete surfaces should be visually inspected. The licensee's requested change would provide for a visual inspection of a random sample of end anchorages and adjacent concrete surfaces during tendon testing consistent with the sample of tendons selected for surveillance, and a sample of tendon-adjacent concrete surfaces during the containment integrated leak rate test (ILRT).

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As indicated in the TS Bases for TS 4.6.1.6.2, the inspection of the containment post-tensioning system (tendon, anchors, and related equipment and structures) is based upon Regulatory Guide (RG) 1.35, "Inservice Surveillance of Ungrouted Tendons in Prestressed Concrete Structures," January 1976. A review of Section C.3 of RG 1.35 clearly indicates that a random sample of tendon end anchorages and adjacent concrete surfaces (corresponding to the random selection of tendons to be tested) should be selected for visual inspection during tendon testing. Thus, the proposed TS change for selection of tendon end anchorages and adjacent concrete surfaces, during tendon testing, is consistent with the provisions of RG 1.35 and is acceptable.

Finally, with regard to observation of concrete surfaces during the containment ILRT test, TS 4.6.1.6.2 requires observation of crack patterns in concrete adjacent to the end anchorages. The licensee proposes to continue the use of a program developed in cooperation with the Architect/Engineer for Calvert Cliffs, Bechtel Power Corporation, as described in BG&E's letter dated June 19, 1985. The program involves the observation of 11 preselected areas for each containment during the ILRT test; each area is 50 to 100 square feet in size. A total of over 50, representative, end anchorages per containment are thus observed. This program has been used at Calvert Cliffs to date. While the RG 1.35 program would incorporate a smaller, random, observation of concrete surfaces (approximately a 1% sample of all tendon-adjacent surfaces), the Calvert Cliffs program involves a larger, fixed, tendon sample (approximately 6% of all tendon-adjacent surfaces per Calvert Cliffs containment). Although the random system of observation might eventually result in a greater range of observed concrete locations, the Calvert Cliffs program incorporates a sufficiently diverse sample to be representative of overall containment concrete conditions.

Based upon the above, the licensee's proposed changes to TS 4.6.1.6.2 and associated Bases, to establish the use of random inspection of tendon end anchorages and adjacent concrete surfaces and the observation of preselected areas during the ILRT tests, is in accordance with RG 1.35, January 1976, and is acceptable.

The licensee has proposed a change to TS 4.6.1.6.3, "Liner Plate". At the present time, TS 4.6.1.6.3 requires periodic visual inspection of the containment liner plate to detect any signs of abnormal degradation. The licensee has proposed to extend this visual inspection to also include ". . . the exposed accessible interior and exterior surfaces of the containment . . . ." This proposed change would improve the visual inspection of the containment and thus increase the likelihood that possible degradation in containment components to other than the liner plate would be detected. In addition, the licensee has proposed a change in the title of TS 4.6.1.6.3 to "Containment Surfaces" to more clearly reflect the nature of the proposed revised visual inspection. Based upon the above, we conclude that the proposed change to TS 4.6.1.6.3 is acceptable.

The licensee has requested a change to TS 4.8.1.1.2c.2 to increase the diesel generator load rejection test load from 450 to 500 hp. The purpose of the load rejection test is to assure that the diesel generator will not trip, due to load rejection, in the event that the electrical load with the highest horsepower rating should trip.

The existing test load specified in TS. 4.8.1.1.2c.2 is 450 hp. Since completion of modifications to the auxiliary feedwater system which added one motor operated pump per unit, the new maximum load is 500 hp. Accordingly, the test load specified in TS 4.8.1.1.2c.2 should be increased to 500 hp to assure that the load rejection test is conducted with the limiting (largest) electrical load.

The proposed change would increase the size of the load that must be periodically rejected by the diesel generator by about 10%. This would provide greater assurance of the generator's capability to respond to the loss of the single largest load and is thus acceptable.

The licensee has proposed to delete TS 3/4.3.3.8 which contains limiting conditions for operation and surveillance requirements for radioactive gaseous effluent monitoring instrumentation. The licensee has further proposed that the requirements of TS 3/4.3.3.8 be incorporated in TS Tables 3.3-6 and 4.3-3 where requirements for similar instrumentation are located. The licensee's proposal to relocate the requirements of TS 3/4.3.3.8 is appropriate since locating requirements for similar instrumentation in common areas within the TS will facilitate compliance and is therefore acceptable.

The licensee has proposed changes to the fire detection instrumentation descriptions contained in TS Table 3.3-11. These instruments are required to be operable and to undergo surveillance in accordance with TS 3/4 3.3.7, "Fire Detection Instrumentation". The proposed changes are of several types as follows:

- One heat detector was replaced with a smoke detector and three more smoke detectors were added as a result of structural modifications to the 69' level access control area. The area includes a laboratory where a smoke detector would be more suitable for fire detection.

- Several duplicate entries occur in TS Table 3.3-11. Both the North South Corridor Room 410 and North South Corridor Room 308 were listed twice. The number of fire detectors in these areas has not been reduced, only the duplicate listings should be eliminated.

- Additional clarification has been proposed as follows: The room numbers and room names should be changed to reflect their proper names. The Intake Structure has been listed as a common structure. Although the Intake Structure is a single room, the equipment in each side is dedicated to its respective unit. To provide clarification, the fire detection instrumentation serving the Unit 1 side of Intake Structure should be exclusively listed in the Unit 1 Technical Specification and similarly for Unit 2.

- The last clarification concerns the Protecto Wire Instrumentation. The existing entries in TS Table 3.3-11 list this instrument location as the Southwest and Northeast Containment Electrical Penetration Rooms. Actually, the instrument meters are located in these rooms, but the Protecto Wires monitor cable trays rather than the rooms themselves. The Protecto Wires are also not conventional heat detectors. If a fire occurs in the cable tray, the insulation between the wires melt and the wires short-circuit. The new electrical resistance corresponds to a wire length which can then be used to determine the location of the fire. A footnote is proposed for TS Table 3.3-11 to clarify the special nature of these detectors.

As noted above, the modification to the fire detection instrument deployment strategy on the 69' level access control area provides a superior degree of fire detection capability. The remaining proposed changes to TS Table 3.3-11 do not in any way impact existing fire detection capability. Thus, we conclude that the overall ability to detect and suppress fires has not been decreased; therefore the proposed changes are acceptable.

The licensee has proposed changes to TS 3/4.6.5 in response to NRC's Generic Letter (GL) 83-37, "NUREG-0737 Technical Specifications", dated November 1, 1983 regarding hydrogen monitors. The hydrogen monitors are required to determine post-LOCA, containment, hydrogen concentrations.

The purpose of GL 83-37 was to provide model TS associated with system/procedural improvements deemed necessary following the accident at Three Mile Island, Unit 2 (TMI-2). The proposed TS change clarifies the Limiting Condition for Operation (LCO) by providing an appropriate remedial action when two hydrogen monitors become inoperable. Although the LCO requires two hydrogen monitors to be operable, the required remedial action is only applicable when one hydrogen monitor is inoperable. The existing LCO allows a single hydrogen monitor to be inoperable for up to 30 days after which the reactor must be shut down within 6 hours. The proposed LCO would require that, when both hydrogen monitors become inoperable, one monitor must be made operable within 72 hours or the reactor must be shut down within 6 hours.

Based upon our review, we conclude that the proposed remedial action, when two hydrogen monitors are inoperable, is consistent with the importance of the subject equipment and is thus, acceptable.

The licensee has also proposed a change to the surveillance requirements for the hydrogen monitors. The proposed change would add a periodic test to the existing calibration requirements of TS 4.6.5.1. The periodic test involves a biweekly demonstration of operability which is performed by drawing and analyzing gas from the waste decay tank. The additional proposed surveillance requirement provides a valid test of system operability at an appropriate frequency. Although the model TS also suggests a more frequent "check" of instrument operability, this type of qualitative observation is meaningless since the hydrogen monitors are maintained in a de-powered state until required.

The proposed changes to TS 3/4.6.5.1 result in increased reliability of the hydrogen monitors in accordance with GL 83-37 and are acceptable.

The licensee has proposed changes to the limiting conditions for operation and surveillance requirements for the Auxiliary Feedwater System (AFW) as specified in TS 3/4.7.1.2. At the present time, the Unit 1 TS 3.7.1.2a.1(b) would allow up to 14 days for a motor-driven AFW pump to be inoperable. In addition, TS 3.7.1.2a.2.(b) allows up to 30 days for a steam-turbine-driven AFW pump to be inoperable. The licensee has proposed that the maximum period of inoperability for either motor-driven or steam-turbine-driven AFW pumps be reduced to 7 days. This proposed change is consistent with the Unit 2 TS.

The proposed change to Unit 1 TS 3.7.1.2 would improve the availability of the Unit 1 AFW pumps by substantially reducing the allowable out-of-service times and is acceptable.

The licensee has also proposed a change to Unit 2 TS 3.7.1.2c which specifies remedial action to be taken when AFW components are inoperable for the purpose of testing. The wording of TS 3.7.1.2c would be changed to allow more than one AFW pump to be inoperable for the purpose of logic testing. For example, testing of the AFW automatic actuation system requires that two of three AFW pumps be momentarily made inoperable. This proposed change is consistent with the Unit 1 TS.

The proposed change to Unit 2 TS 3.7.1.2c would only insignificantly decrease the availability of the AFW system. Moreover, the existing TS 3.7.1.2c requires a dedicated operator to be stationed at the AFW pumps (with direct communication to the control room) to promptly restore full AFW capability in the event of an accident. The proposed change to TS 3.7.1.2c is therefore acceptable.

The licensee has proposed the following changes to the Unit 1 and Unit 2 TS 3/4.7.1.2:

- Delete the note addressing Unit 1, Cycle 7, system inoperability. This note is no longer applicable.
- Correct the spelling of "standby" in a note in the Unit 2 TS. This change would correct a typographical error.
- Correct the spelling of "characteristics" in the Unit 1 TS. This change would correct a typographical error.
- Add the word "and" to a Unit 1 surveillance requirement. This change would correct a clerical error.
- Add a close parenthesis to a Unit 2 surveillance requirement. This change would correct a clerical error.

These proposed changes are minor in nature and do not affect the AFW system or related analyses and are 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 and 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 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). 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 9, 1985

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